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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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ONGOING AND PLANNED FUEL SAFETY RESEARCH IN NEA MEMBER STATES

Compiled from SEGFSM Members' Contributions

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14th December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

Abstract

This report is in response to an action placed on SEGFSM members to compile ongoing and planned fuel safety research in NEA member states with the aim of providing CSNI an overview on related R & D international programmes and projects, along with the identification of current and future needs and priorities.

The report is based on replies to a questionnaire distributed to SEGFSM members requesting them to identify fuel safety research programmes and to provide information on achievements and future plans. The report is confined to the replies received, as a consequence it cannot be viewed as comprehensive; programmes may well be in progress in addition to those detailed here.

The report is organized in topic sections relating to: fuel and clad studies, integral fuel rod tests and PIE, LOCA and RIA studies including whole rods and bundles as well as single effects studies of fuel and cladding, code development for both steady state and transient fuel behaviour, thermal hydraulics, reactor physics codes and finally severe accident studies.

Fuel Safety Research in NEA Member States

Executive Summary

This report is in response to an action placed on SEGFSM members to compile ongoing and planned fuel safety research in NEA member states with the aim of providing CSNI an overview on related R & D international programmes and projects, along with the identification of current and future needs and priorities.

A questionnaire was distributed to SEGFSM members on 18 October 2000, requesting them to identify fuel safety research programmes and to provide information on achievements and future plans. The questionnaire required respondents to provide information on the ongoing R&D programmes under the following headings:

A. Title
B. Research Laboratory/Sponsor(s)
C. Objectives/Goals
D. Status of Work
E. Brief description/presentation of the main results achieved
F. Future plans
G. References

Replies were received from organizations in the following countries:

- Belgium
- Canada
- Czech Republic
- France
- Germany
- Hungary
- Japan
- Korea
- Norway (Halden Reactor Project)
- Sweden
- Switzerland
- United Kingdom
- USA.

The report is based on the information provided in the replies received, as a consequence it cannot be viewed as comprehensive; programmes may well be in progress in addition to those detailed here. It is also possible that the detailed results of some programmes may remain proprietary and therefore not available in the short term.

The report is organized in topic sections relating to: fuel and clad studies, integral fuel rod tests and PIE, LOCA and RIA studies including whole rods and bundles as well as single effects studies of fuel and cladding, code development for both steady state and transient fuel behaviour, thermal hydraulics, reactor physics codes and finally severe accident studies.

The main issues for the current generation of reactors are those of high burn-up performance in normal operations, LOCA and RIA conditions and the main goal for the industry is to consolidate the safety issues to bring all countries up to a licensed discharge burn-up of ~60 MWd/kg in assembly average and possibly 65 MWd/kg. The principal issues requiring attention can be broken down as follows:

Normal operation:

- fission gas release and rod over-pressure,
- properties of the High Burn-up Structure (HBS) at the pellet rim, its effect on thermal performance and fission gas release,
- cladding oxidation, hydriding and embrittlement.

Loss of Coolant Accidents (LOCA)

- the possibility of fuel slumping into the ballooned region; the effect of fuel-clad bonding at high burn-up,
- increase of pressure in the ballooned region due to fission gas release from slumped fuel,
- response of irradiation hardened and hydride embrittled cladding.
- review of the 17% Equivalent Clad Reacted (ECR) criterion,

Reactivity Initiated Accidents (RIA)

- PCMI (Pellet-Clad Mechanical Interaction) loading mechanism(s) on the cladding, the effect of the HBS at the pellet rim,
- effect of HBS on fuel dispersal,
- response of embrittled cladding to transient PCMI.

In addition to these, for those countries that load both MOX and UO_2 fuelled assemblies, there is a requirement to bring the MOX database to the same level as that for UO_2 fuel with the aim of treating MOX indistinguishably from that of UO_2 as far as safety is concerned.

The survey of international research programmes outlined in the report demonstrates the large element of activity to address these issues. When put together, the individual programmes add up to a tremendous effort in both time and money and will ultimately lead to a much better understanding of materials and component behaviour in a wide range of postulated scenarios. It is to be noted that all countries have extensive modelling and code development programmes to best utilize the data generated from the experimental programmes.

It is very important therefore, that these activities are well supported and that their results should be made available to the widest possible audience. Only in this way can there be a global common culture of safe and economic production of electricity from nuclear power generation.

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1. Introduction

This report is in response to an action placed on SEGFSM members to compile ongoing and planned fuel safety research in NEA member states with the aim of providing CSNI an overview on related R & D international programmes and projects, along with the identification of current and future needs and priorities.

The report is based on replies to a questionnaire distributed to SEGFSM members requesting them to identify fuel safety research programmes and to provide information in the following form:

A. Title
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The report is confined to the replies received, as a consequence it cannot be viewed as comprehensive; programmes may well be in progress in addition to those detailed here. The report is organized in topic sections as outlined below. Information from each country and laboratory is included within each topic area.

The main areas of research are concerned with studies on and relating to LOCA and RIA. These areas are identified and treated separately in sections 5 and 6 respectively. In addition to prototypic or simulation tests, the appropriate sections are divided to address separate effects experiments on both fuel and cladding. The report starts with section 2 on fuel studies and section 3 on cladding programmes not explicitly concerned with these accident scenarios. Section 4 deals with integral fuel rod tests. Here are included prototypic irradiations and ongoing PIE programmes. Also in this section is an account of dry-out tests and associated PIE performed in the OECD Halden Reactor Joint Programme. Fuel performance code development for both steady state and transient applications is being pursued in many countries, and these activities are reported in section 7. Section 8 covers thermal hydraulic studies, section 9 the development of reactor codes in Japan and finally, section 10 covers molten fuel-moderator interaction tests performed in Canada, under the generic title of Severe Accident Studies.

2. Fuel Studies

2.1 UO_2

2.1.1 Chalk River Laboratories (CRL) Hot-Cell Fission-Product Release Experiments, (Canada)

These hot-cell fission-product release experiments were performed at the Chalk River Laboratories of AECL. The CANDU Owners Group (COG) sponsored these experiments under joint funding from Ontario Power Generation, Hydro Québec, New Brunswick Power, and AECL.

The CRL hot-cell fission-product release experiments were performed to improve our understanding of fission-product release from CANDU fuel under accident conditions, and to provide data for use in reactor safety code validation. Over 250 tests of fission-product release from clad and unclad CANDU fuel samples under various CANDU accident conditions have been performed.

The results obtained in the investigation of fission-product releases from unclad UO_2 fuel samples between 1983 and 1992 were summarized by Z. Liu, et al. [1]. The results of these experiments allowed the fission products to be grouped into the following categories according to their release behaviour: noble gases (Xe and Kr), volatiles (Cs, I, Te and Tc), “semi-volatiles” (Ru, Nb and Sb), and low-volatiles (Ba, Zr, La, Ce, Pr, Eu, Nd and Pm). The noble gases and volatiles are released rapidly when the fuel is heated to high temperatures. The “semi-volatiles” are potentially volatile depending on the environmental conditions. Ru and Sb releases are observed at relatively low temperatures under oxidizing conditions. Nb is released at lower temperatures in steam, than in air or inert atmospheres. The low-volatile fission products are released mainly by matrix stripping, i.e., by volatilization of the UO_2 itself [2,3]. Recent studies have focused on the effect of atmosphere (inert, steam and air) on releases from clad fuel samples [4], the effect of rapid sample heating rate on releases from clad fuel samples in the presence of a large fuel temperature gradient [5], and measurement of the fission-product grain-boundary inventories in CANDU fuel [6].

Fission-product release from Zircaloy-clad segments of CANDU fuel was measured in six tests conducted in argon, steam and air environments at peak temperatures between 1780 and 2200 K [4]. On-line gamma-spectrometry showed significant fractional releases for Xe, Kr, I, Cs, Te and Ru. Post-test deposition measurements also detected releases of Sn, Sb, Ag, Ba, Zr and Nb. In the oxidizing atmosphere tests, the release rates of Kr, Xe, I and Cs were very low prior to complete oxidation of the cladding. The peak release rates of the noble gases and volatiles in steam after complete clad oxidation were not very temperature-dependent. Ru release was delayed by more than 2000 s after oxidative releases of the volatile fission products.

A direct-electric-heating apparatus was developed to heat Zircaloy-clad irradiated fuel samples [5]. The apparatus was used to measure fission-product releases from Zircaloy-clad CANDU fuel samples during fast temperature ramps in the presence of a radial temperature gradient in the fuel. The ohmic heating of the UO_2, combined with surface heat removal by the surrounding helium coolant flow, produced a radial temperature profile that approximates the profile for fission- or decay-heated fuel. Ceramographic examinations showed columnar grain growth and evidence of UO_2 melting. Releases of Kr, and release and redistribution of Cs were observed. The amount of Kr released was highly dependent on the peak dwell power: the higher the dwell power, the higher the cumulative release. Cesium was released from the centre of the fuel sample where the temperatures were the highest. The measured Kr releases were in good agreement with the Cs migration and release.

The Kr-85 grain-boundary inventory (GBI) was measured for ten CANDU power reactor fuel rods, with peak linear power ratings between 26 and 58 kW/m [6]. Samples were cut from central, mid-radial and peripheral locations in the fuel pellets. Peripheral GBI values were less than 5%, while the GBI of most central samples was less than 50% of the Kr-85 remaining in the sample, giving volume-average GBI values in the range 1% to 9%. Some Kr-85 in large intragranular bubbles was probably also released during the initial oxidation step by cracking of the grains during oxidation to U_3O_8. The results will be used in validation of fission-product distribution models in steady-state fuel performance (ELESTRES-IST) and transient fission-product release (SOURCE-IST 2.0) codes.

Future studies will focus on the effect of heating rate and the effect of hydrogen-rich atmospheres on fission-product release from CANDU fuel.

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2.1.2 Transient Behaviour of High Burn-up UO_2 Rim Structure, (Korea)

The Korean Atomic Energy Research Institute have initiated a programme to study kinetics of fission gas release, bubble swelling and microstructure transformation of high burnup UO_2 rim structure during thermal transient. At present the programme is focussing on developing test equipment to simulate a thermal transient and it is anticipated that tests will commence in 2002.

2.1.3 PSI PIE Programme for High Burnup Fuel, (Switzerland)

Experimental fuel research takes advantage of the fact that two modern LWR's (BWR-KKL and PWR-KKG) with high linear heat ratings have ambitious HBU programmes for UO2 fuel and in the case of KKG also for MOX fuel. After pool side characterization, fuel pins form the mentioned power stations are sent to PSI for detailed NDT and destructive testing. The latter includes Cladding-TEM, -EPMA and -SIMS characterization, and fuel-EPMA and -SIMS studies (e.g. isotope distributions in the fuel rim region). The same equipment will also be used for fission gas analysis in the rim region, to define the starting conditions for hypothetical LOCA and RIA transients. The idea is to repeat these analyses later, after completion of fuel transients on neighbouring samples. A close connection with modelling is established as the experimental results will be used to optimize the fission gas model in the computational methods.

The experimental safety related fuel research has been performed so far on UO2 fuel up to 80 MWd/kg and for MOX fuel up to 55 MWd/kg. The PSI possibilities to characterize the isotopic distributions of fissile and fission products across fuel polished sections are unique.

The involvement in the characterization of HBU-fuel will continue. PSI will prioritize fission gas analysis in the high burnup structure. As far as fission product speciation in fuel is concerned, it is anticipated that an EXAFS beam line will be introduced at the PSI synchrotron light source. The results will be used to develop or validate new models in the fuel codes.

2.2 MOX

2.2.1 BNFL MOX Fuel Development Programme, (UK)

BNFL manufactures MOX fuel using the Short Binderless Route (SBR) at its Sellafield site. A major development programme is being undertaken to qualify SBR MOX fuel for LWR commercial reactor operation through to high burnup. The programme covers microstructural characterisation, property measurement work, test reactor irradiation, ramp testing, post-irradiation examination and various specialised post-irradiation studies, together with mathematical modelling. Experimental data for this research comes mainly from BNFL's own programmes, although use is also made of international collaborative projects. These include the general programmes in Halden, NFIR and CABRI, plus MOX-specific projects such as Callisto, M308 and MicroMOX. The main ingredients of the research are as follows:-

Material characterisation and physical property measurements
Comprehensive characterisation of unirradiated SBR MOX fuel has resulted in detailed knowledge of the microstructure and homogeneity of the product. The EDS mapping technique shows that typically less than two percent of SBR fuel has plutonium concentration above twenty weight percent. These maps allow a detailed study of the distribution of spot sizes and show over 95 percent of the spots have an equivalent diameter less than 20 microns. The already extensive knowledge of the unirradiated product is being added to with the development of further investigative techniques. These include the use of SIMS mapping on both unirradiated and irradiated fuel, which expands on the work of plutonium mapping via EDS techniques. SIMS allows the determination of the distribution of individual isotopes and has greater resolution than x-ray micro-beam analysis. Other work includes the characterisation of porosity distributions in as-fabricated fuel using SEM techniques. This quantifies the level of sub-micron porosity in fresh fuel, which correlates closely with the degree of pellet densification during the first stages of irradiation. BNFL has also performed a significant amount of work determining the fundamental physical properties of SBR fuel, which generally confirm the previously published data on other MOX fuels. There are continual improvements in some measurement techniques and hence, in particular, further melting point measurements are planned to obtain an improved determination of this fundamental parameter.

In-pile testing
Successful in-pile testing is a vital step in obtaining confidence in the irradiation performance of a new fuel type. One Halden bilateral experiment has already successfully irradiated small diameter SBR MOX fuel pellets to 80 MWd/kgHM, while three other bilateral rigs containing prototypic PWR and BWR fuel designs are also now reaching high burnups. An intermediate PIE programme has examined fuel from one of these experiments at 28 MWd/kgHM and revealed no items of concern. A Halden joint programme experiment, containing an instrumented SBR rod, has recently begun in the framework of the inert matrix fuel testing project. There is also the widely reported BNFL gas flow experiment IFA-633. Useful information has already been gleaned from this experiment on beginning of life thermal performance, gap conductance and stable and unstable fission product release. The PCI (pellet-clad interaction) resistance of MOX fuel is generally found to be superior to that of UO2 fuel. This has been confirmed for SBR fuel in a small ramp test programme as part of the Callisto project, while a definitive ramp test programme using commercially irradiated fuel segments is due to commence during 2001.

Post-irradiation studies
SBR MOX fuel has been irradiated in European commercial reactors since 1994. In 1998, after three cycles of irradiation, seven rods were taken for PIE from the M5 region in Beznau-1. This extensive programme (M501) has been widely documented and is now effectively complete. Higher burnup data is now needed and a PIE programme has begun on four cycle M5 region fuel. The rods are at the crucial range of burnup where acceleration of fission gas release has been observed in other MOX fuels.

A future programme is planned on five cycle fuel, which will provide data above 50 MWd/kgHM. Additional studies include solubility testing, which is underway, and a schedule of post irradiation anneals, which are part of a long-term programme to validate fission gas release modelling.

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2.2.2 NUPEC-MOX Irradiation Performance Programme, (Japan)

Some fuel assemblies, out of the ones which will be irradiated in commercial BWR and PWR in Japan, will be supplied to the NUPEC/METI sponsored programme for PIE. The PIE will be performed in the NFD, (Nippon Fuel Development) and NDC, (Nuclear Development Corporation). The objective of this project is to collect MOX fuel performance data to confirm the integrity and performance of the MOX fuel irradiated in commercial power reactors. Characteristic data will be collected in each step of fabrication, irradiation and post-irradiation in order to obtain systematic and reliable data.

To date, a part of MOX fuels for irradiation have been fabricated, the pellets have been characterized and the Pu homogeneity investigated by alpha-autoradiography and EPMA. Also, the melting temperature and thermal conductivity, etc. of other irradiated MOX pellet has been measured.

BWR type MOX fuel will be irradiated up to approximately 40 GWd/t (fuel assembly average) at a commercial NPP in Japan. The PIE of the 1 to 3-cycle irradiated fuel assemblies will be carried out. PWR type MOX fuel will be irradiated up to approximately 45 GWd/t (fuel assembly average) at a commercial NPP in Japan. The PIE of 2 and 3-cycle irradiated fuel assemblies will be carried out.

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2.2.3 BELGONUCLEAIRE MOX R &D Programme, (Belgium)

BELGONUCLEAIRE, pioneer of MOX fuel for LWRs and FBRs, fabricates today MOX fuel following the MIMAS process developed by the company in the years 80. The process is used now in France as well and is contemplated for future MOX production in Japan and USA.

The in-reactor performance of BELGONUCLEAIRE fuel has been assessed since a long time through experimental data obtained in the framework of R&D programmes supported by the European Community and the Belgian State at the very beginning and, today, by auto-financing and International Programmes managed by BELGONUCLEAIRE, often in collaboration with the Belgian national nuclear centre (SCK.CEN), and supported by numerous companies (fuel manufacturers, designers and vendors, utilities, safety authorities, research centres) interested in MOX in-reactor performance.

The activities covered by an International Programme include the fuel fabrication and its characterization, its irradiation in commercial LWRs where burnup is accumulated and in MTRs where transient tests are performed, and post-irradiation hot cell work for fuel characterization and complementary testing.

MOX fuels examined in the BELGONUCLEAIRE programmes approach 80 GWd/tM peak pellet burnup.

During the examinations, attention is paid to the MOX performance in all its aspects: fuel rod integrity ; cladding corrosion, deformation, hydrogen up-take, short and long term mechanical properties ; extent of pellet-cladding mechanical; rod inner pressure and fuel fission gas release; fuel microstructural features, physical properties and chemical inventory.

These characterizations allow to draw general conclusions on MOX performance and to compare it to UO2 one.

MOX threshold for rod failure in power ramp (100 W/cm min) is higher than UO2, due to its lower creep strength. Regarding PCMI in steady state conditions, MOX is however similar to UO2.

Extensive work is made on fission gas release. Indeed, surveillance programmes show that it is more important than in UO2 at the same burnup. The exact origin of this enhancement is still the object of some speculation and no definitive explanation has been given so far. For this reason, R&D is pursued in this field, together with studies on high burnup MOX structure and helium production and impact on performance.

3. Clad Studies

3.1 Mechanical Properties

3.1.1 Mechanical Properties of High Burn-up Fuel Cladding, (Korea)

The Korea Atomic Energy Research Institute has initiated a programme to measure the mechanical properties of cladding at high burnup. The programme is at an early stage with test methods for the irradiated cladding under development and qualification. Measurement of the irradiated cladding mechanical properties by the qualified methods will be started from 2002.

3.1.2 Mechanical Testing of Zr1%Nb Cladding with Different Oxygen and Hydrogen Content, (Hungary)

The KFKI Atomic Energy Research Institute of Hungary, sponsored by the Hungarian Atomic Energy Authority National Committee for Technological Development has started this programme using unirradiated cladding. The objective of the study is to determine separately the effect of O and H content on the mechanical properties. The series of tests completed investigated the range covering 0-2400 ppm H content and 0-200 μm oxide layer thickness. The measurements were carried out between 20 and 300°C.

The results showed that the H content is more responsible for the degradation of mechanical properties than the O content. The strength of the cladding had a local maximum between 2-4% oxidation. Future plans are to continue the test series with irradiated cladding.

Further work performed by KFKI Atomic Energy Research Institute on clad mechanical properties in section 5.2 under LOCA separate effects testing.

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3.1.3 Incomplete PWR Control Rod Insertion (IRI), (Sweden)

The objective of this project is to study the in-pile creep of modern PWR guide tube materials under a constant bending moment. Research is carried out by the Studsvik Nuclear and sponsored by both the SKI and the Swedish Nuclear Industry Research Programme/Fuel Group (BFUK/PGB).

In-pile creep testing of initially unirradiated specimens of two commercial tube materials has been almost completed. Pre-irradiation without load up to a flux of $1 \cdot 10^{21}$ n/cm^2 (E>1 MeV) has been completed and in-pile and thermal creep testing of the pre-irradiated specimens will begin late 2002. The pilot study using a non-commercial Zircaloy-4 tubing demonstrated the testing method and showed that the initial creep rate in-pile is substantially higher than the thermal creep at the same temperature. In the future in-pile creep testing of the pre-irradiated tubes will be performed.

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3.2 Corrosion Behaviour

3.2.1 Kinetics of the H_2 Absorption and Desorption of Zr Alloys, (Hungary)

The National Committee for Technological Development also sponsored this programme which KFKI Atomic Energy Research Institute has now completed.

The research project aimed at the determination of the H_2 source term at high temperature steam oxidation which is not only depending on the generated hydrogen but on the absorption/desorption phenomena, as well. The most important result of the experimental program is the determination of the Arrhenius type correlation considering the effect of oxide layer thickness on the process. No further work is planned.

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3.2.2 Out of Pile Experimental Programme on Corrosion of Zr-Alloys, (Czech Republic)

This programme is being conducted by SKODA-UJP of Prague-Zbraslav, sponsored by CEZ Prague and Czech Ministry for Industry and Trade. The objectives are to study the long-term corrosion behaviour of different Zr-alloys in water and steam environments and to correlate the behaviour of the alloys with the properties of the corrosion layer. A particular attention is paid to the results in VVER imitating water. This extensive out-of-pile experimental program started in 1995 and has continued over the subsequent five years.

The specimens of four Zr-alloys were simultaneously exposed in pressurized autoclaves:

- Low-Tin Zircaloy-4
- ZIRLO^{TM}
- Standard Zircaloy-4
- Standard VVER Zr1Nb.

Water with different chemistry and steam were selected as the experimental environment, and the following 360°C water chemistries were tested:
VVER-1000 imitating chemistry

- 70 ppm and 210 ppm Li as LiOH
- 43 ppm O_2
- 450 ppm Li as LiOH + 17,5 ppm B as H_3BO_3

with different exposure times up to ~1080 days. For exposure in 400, 450 and 500°C steam at ~10MPa the time was up to ~1000 days.

The weight gain and axial elongation were measured on all specimens, the absorbed hydrogen on selected specimens. Oxide scales were studied in detail by optical metallography, scanning electron microscopy, atomic force microscopy, and photoelectron spectrometry. The oxide resistivity was also measured. The stress in the oxide and underlying metal was determined by RTG diffraction tensometry, the fraction of tetragonal oxide by Raman spectroscopy, the permeability of the oxide by electrochemical impedance spectrometry and the porosity by high-pressure mercury porosimetry.

Although the longest planned expositions have not yet been achieved, the results available allow the following summary:

- The corrosion kinetics and hydridation of Zr1Nb and ZIRLO^{TM} in the pre-transition stage depends on the corrosion environment (for the same temperature), but does not depend in case of Zircaloy except the varying time at transition. After transition, however, the corrosion environment influences the behaviour of all alloys.
- The hydridation of alloys in lithium-containing water (compared to VVER water) differs from each other. Zircaloys after transition absorb with increasing Li-content much more hydrogen (>20%) than Zr1Nb and ZIRLO^{TM} (<5%).
- The results concerning the development of the tetragonal phase in Zr-alloys have lead to a hypothesis that the good corrosion properties of Zr1Nb and ZIRLO^{TM} can be among others (and in contrast to Zircaloy) associated with the stabilization of this phase due to irradiation (influencing the delay of transition).

Studies of the corrosion layers will continue over the next 3 years on specimens with longer exposures up to the termination of the project. In case of enlarging the experimental studies to the newly developed Zr-alloys the program will continue for an additional period.

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3.2.3 PSI PIE Programme, (Switzerland)

With respect to cladding, emphasis was placed on detailed characterization of corrosion layers (e.g. lithium distribution in PWR oxides) HR-TEM to characterize the metal-oxide interface, hydrogen distribution between Oxide and Metal. The radiation- and hydride embrittlement effect was evaluated by tensile and burst testing.

4. Integral Fuel Rod Tests

4.1 Prototypic Irradiation and PIE

4.1.1 OECD Halden Project Joint Programme, (Norway)

OECD Halden Reactor Project's joint research and development programme is sponsored by 20 participating countries. The framework of the programme is defined for a period of three years, currently from 2000 to 2002.

The objective of the joint programme is to provide performance data of fuel and cladding variants under normal and transient operation conditions with emphasis on high burnup effects. Phenomena and issues related to extended burnup under normal operation conditions include:

- Thermal performance and degradation of fuel thermal conductivity with increasing burnup
- Pellet-cladding interaction due to fuel swelling
- Rod pressure due to fission gas release
- Cladding corrosion and hydrogen pickup
- Cladding creep properties at extended exposures
- Tolerable rod pressure limits

While UO_2 continues to be addressed extensively, knowledge on MOX fuel behaviour is also being acquired. In addition, gadolinia fuel and VVER fuel are used in some tests. The majority of the investigations involve the extended irradiation of test and commercial fuel which already has significant burnup (60 - 100 MWd/kg). For the latter, the well-proven technology for re-instrumenting with sensors for measurement of fuel temperature, rod pressure and rod length changes is utilised.

Among others, the normal operation data are required for the proper definition of initial conditions prior to a transient.
Many tests, especially when related to high burnup, extend over several years and are on-going. The results are used by participants for their code development and validation efforts.
The results of these studies are too varied for a comprehensive summary. In addition to results of a recent Dry-out test reported in section 4.2, a few more general observations are made below:

- *Clad creep:* A number of cladding materials have been and are being studies with respect to their creep behaviour under varying stress conditions. Primary and secondary creep have been investigated as a function of stress change and stress level, and the data allow assessing the fuel-clad gap as well as the state of PCMI.
- *Clad overpressure / lift-off:* The overpressure required for outwards creep in excess of fuel swelling (gap opening) has been investigated with different combinations of fuels (UO_2, MOX) and cladding. Pressures well beyond the system pressure are required to produce increasing fuel temperatures.

- *Fission gas release:* The rod inner pressure is related to the lift-off issue as well as to the ballooning behaviour during a LOCA. Several fuel types (UO_2, MOX, U-GdO_2) with low to high burnup are being used in on-going studies. Compared to standard modelling of fission gas release, a release enhancement is found for high burnup fuel. The release of unstable isotopes is assessed with gamma spectrometry, and the results are being used for updating the ANS 5.4 FGR model for application to high burnup fuel.
- *Axial gas flow:* Invariably, an impeded axial gas communication is found for high burnup fuel – especially at power, but also at hot stand-by. In full-length commercial fuel rods, the gas flow will be even more restricted than in the shorter (40 – 50 cm) test segments. Axial gas communication has a bearing on LOCA behaviour (mass transport to the ballooning spot).
- *Thermal properties:* Good temperature predictions remain essential for the analysis of fuel behaviour under both normal and transient conditions. Many experiments have provided information on fuel conductivity and its change with burnup (UO_2, MOX, U-GdO_2).

The future plan for experimental work is based on recommendations from member organisations, on discussions in the Halden Board and Programme Group, and on advice from specialist workshops. The definition of a continued programme is in progress. A major consideration is to provide data supporting a mechanistic understanding of potential life-limiting phenomena focusing on representative and modern fuels and materials. Destructive and non-destructive PIE will be included in the programme.

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4.1.2 BWR Fuel Irradiation Test, NUPEC-HB-B Project, (Japan)

This a collaborative programme of work between Research laboratories and hot cells of NFD, and JAERI, with irradiation ramp testing performed in the JMTR of JAERI in Japan. The sponsors for this work are NUPEC/METI (Ministry of Economy, Trade and Industry).

The objective of this project is to collect higher burnup fuel performance data to confirm the integrity and performance of new fuel design for higher burn-up, by using lead-use assemblies, (LUAs) irradiated in commercial power reactors. Characteristic data on fuel are collected in each step of fabrication, irradiation, post-irradiation, and power ramp test order to obtain systematic and high quality data and to confirm the effect of design improvements.

Burn-up extension is proceeding step by step in Japan. The target bundle maximum burn-up of BWR fuels in Japan are 40 GWd/t for Step 1 fuel, 50 GWd/t for Step 2 fuel

(8x8), and 55 GWd/t for Step 3 fuel (9x9). (See Table 4.1.1) All the PIE on Step 1 fuel have been completed. Irradiation, ramp tests and all PIE of Step 2 fuel have been completed . Step 3 fuels are under irradiation for the fifth cycle in a commercial power reactor. Two LUAs of Step 3 fuel after 3 cycles of irradiation are under PIE in hot laboratories.

Step 2 LUA has achieved 47.8 GWd/t bundle average burnup and showed a good performance. Base irradiation results include:

- The soundness of the fuel assemblies was confirmed up to 47.8 GWd/t.
- FGR of Step 2 LUA was less than those of previous type of fuels due to design improvement to reduce fuel pellet temperature, Fig. 4.1.1.
- Water-side corrosion of fuel cladding tube of Step 2 LUA was substantially improved due to the use of corrosion resistant Zircaloy-2 cladding tube, Fig. 4.1.2.
- Hydrogen pick-up of fuel cladding tube of Step 2 LUA has been low up to 3 cycle, but was increased at 4 cycle, Fig. 4.1.3.

The results of power ramp tests show:

- The failure threshold power decreases at high burnup, but there is still sufficient margin compared to the maximum design limit, Fig. 4.1.4.
- The cracks initiated from the outer side of cladding on high burnup rods were caused by radial precipitated hydrides and its propagation mechanism seem to be DHC(delayed hydride cracking) different from that of PCI/SCC.

The project for Step 2 LUA will be completed in March 2002.

For Step 3 LUA, irradiation up to five cycles will complete in March 2003. PIE on 3 and 5-cycle irradiated LUAs will be completed by the end of the project in March 2006.

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Table 4.1.1 Comparison between 8x8 (Step 2) and 9x9 (Step 3)

Main specification	8x8	9x9 type A	9x9 type B
1.Fuel bundle Number of rod Max. bundle burnup (GWd/t)	 60 50	 74 (include partial length rod 8) 55	 72 55
2.Water rod Number of rod Shape of cross section	 1 Circle	 2 Circle	 1 Square

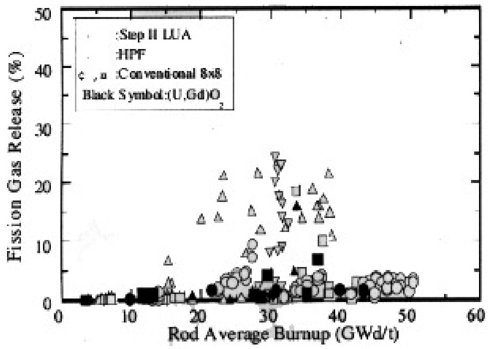

Fig. 4.1.1 Burnup Dependence of Fission Gas Release

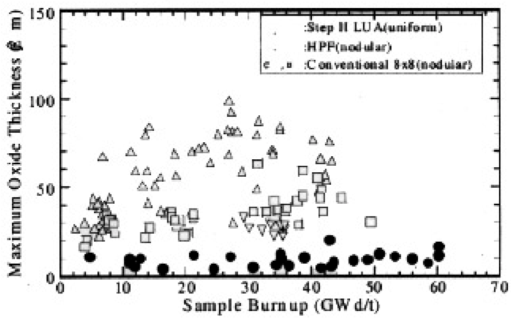

Fig. 4.1.2 Burnup Dependence of Maximum Oxide Thickness

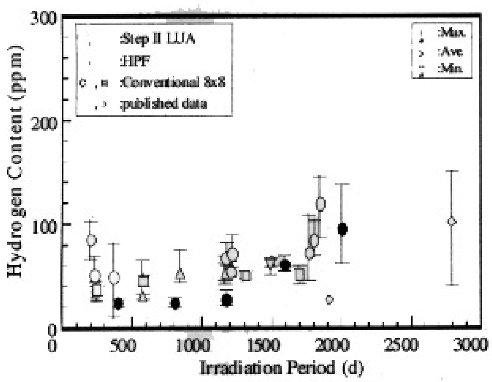

Fig. 4.1.3 Hydrogen Content Variation with Irradiation Period

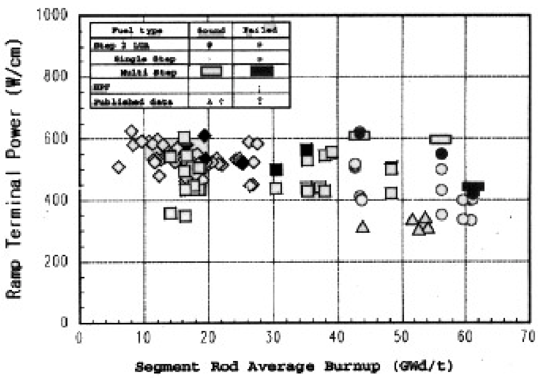

Fig. 4.1.4 Results of Power Ramp Test for Zr-lined BWR Fuels

4.1.3 PWR Fuel Irradiation Test, NUPEC-HB-P Project, (Japan)

This is a collaborative programme of work between Research laboratories and hot cells of JAERI and NDC in Japan, Studsvik in Sweden and Kjeller in Norway. Base irradiations and on-site inspection is performed in Vandellos, Spain, ramp testing performed in R2 of Studsvik, Sweden and other irradiations and ramp testing in the HBWR at Halden, Norway. The sponsors for this work are NUPEC/METI (Ministry of Economy, Trade and Industry).

The objective of this project is to confirm the integrity and to collect performance data on high burn-up PWR Fuel Assemblies, See Table 4.1.2. To do this, PIE is conducted for 48GWd/t fuel (STEP-1,Low-tin Zircaloy-4 cladding fuel) irradiated in commercial power reactors in Japan. PIE of 55GWd/t fuel (STEP-2, advanced cladding with controlled texture and large grain size pellet) has also been performed.

All tests on STEP-1 and STEP-2 fuel have been completed.

Results to date are as follows:

< Step-1>

- The result of PIE on a 43 GWd/t Step-1 fuel assembly showed integrity and high burn-up performance characteristics.

< Step-2>

- Waterside corrosion of low-tin Zry-4 tends to be accelerated at high burnup, while this tendency is not observed in the advanced claddings, Fig 4.1.5 .
- Hydrogen content increases with oxide thickness. Hydrogenpickup ratios are between 10 and 20 %, Fig 4.1.6 .
- The result of FGR measurements confirmed the effect of large grain size on the FGR of pellet, Fig 4.1.7.
- In Power Ramp Tests, the segmented rods are intact well above the conventional PCI failure threshold and sufficient margin against the threshold value is confirmed, Fig 4.1.8.

Table 4.1.2 Comparison between Type A and Type B fuels.

	STEP-1		STEP-2	
	Type A	Type B	Type A	Type B
Cladding	Normal-tin Zry-4 Low-tin Zry-4	Normal-tin Zry-4 Low-tin Zry-4	Low-tin Zry-4 MDA ZIRLO	Low-tin Zry-4 NDA
Pellet	Normal grain size	Normal grain size	Normal grain size Large grain size	Normal grain size Large grain size
Fuel Assembly	Normal assembly Bottom-off	Normal assembly Bottom-on	Assembly with segmented fuel rods	Irradiation rig with short length fuel rods

Type-A is made by Mitsubishi Heavy Industry.

Type-B is made by Nuclear Fuel Industry.

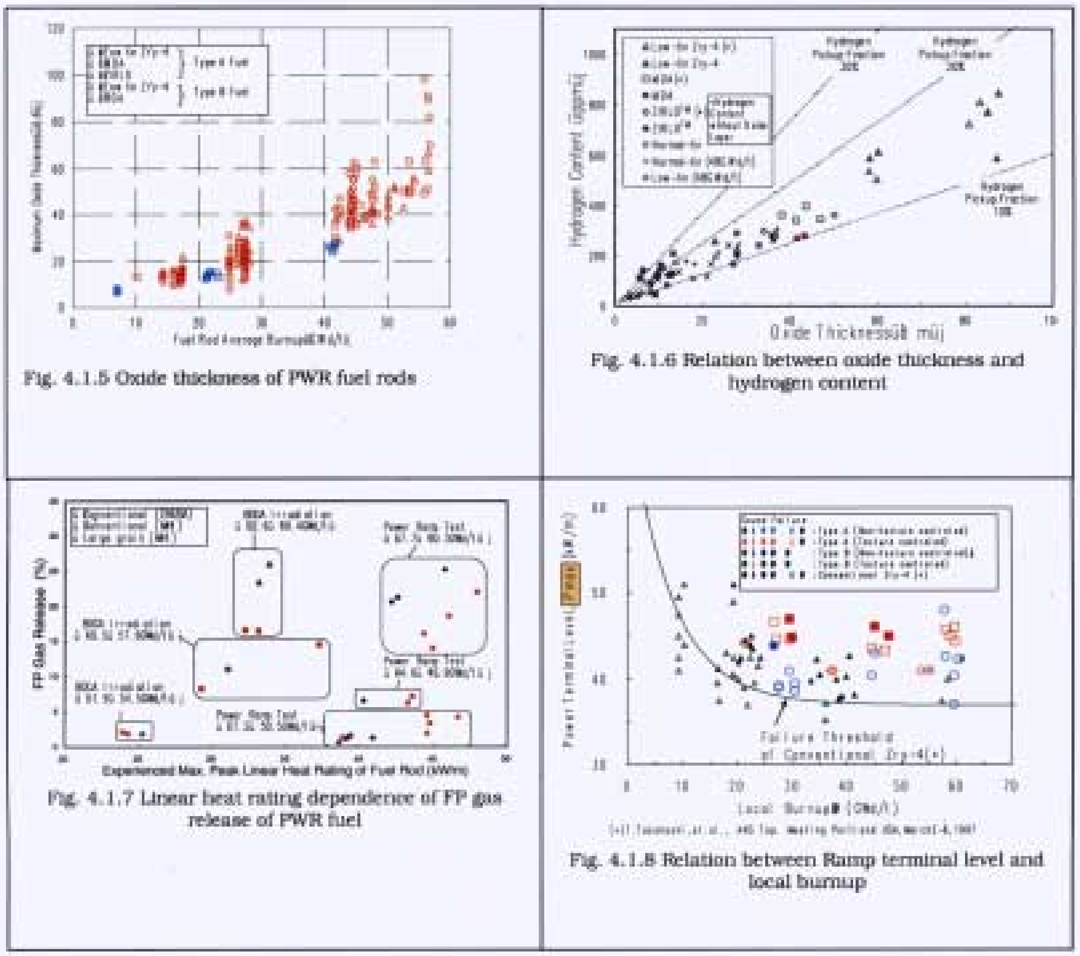

Fig. 4.1.5 Oxide thickness of PWR fuel rods

Fig. 4.1.6 Relation between oxide thickness and hydrogen content

Fig. 4.1.7 Linear heat rating dependence of FP gas release of PWR fuel

Fig. 4.1.8 Relation between Ramp terminal level and local burnup

4.2 Off-Normal Experiments

4.2.1 OECD Halden Project Dry-out Effects on High Burn-up Fuel, (Norway)

The test series was co-sponsored by the OECD Halden Reactor Project's Joint Programme and TEPCO (Japan).

The objective of the dry-out test series was to provide information on the consequences of short-term dry-out incidents in a BWR for the fuel. The experimental method employed was, on an individual basis, to expose fuel rods with different burn-ups to single or multiple dry-out events; to follow this by either unloading or continued operation in the reactor; and to finish with post irradiation examination and testing with emphasis on fuel clad properties.

The test series comprised three loadings of IFA-613. Each rod was contained in a stainless steel channel within the rig so that the coolant conditions for each rod could be controlled individually. In this way separate dry-out scenarios were effected for each rod. Thermocouples attached to the surface of the test rods were used to monitor clad surface temperature and clad elongation was monitored by way of an extensometer. The first and second loading operated for a month after dry-out whilst the rods in the last loading were unloaded directly after the dry-out procedure. In neither case did any fuel failures develop.

The in-pile dry-out experiments with the third (and last) set of fuel rods in IFA-613 were completed in January '98 (HWR-552, HP-1036) and the post irradiation examination (PIE) on all eight rods in the three test series were finished in September '98 (Kjeller hot cell).

In total, 2 rods with fresh Zr-2 and Zr-4 and 6 rods with clad pre-irradiated to 22-40 MWd/ kg (Zr-2, Zr-2 with liner and Zr-4) were individually exposed to reduced or no-flow conditions in a heated light water loop within the Halden reactor. Dry-out occurred over the upper region of each rod, with 6 rods developing peak clad temperatures in the range 950-1200 C occurred in the other two rods.

An overview of the condition of the rods in terms of clad surface condition, rod dimensions and hydriding was achieved using non-destructive PIE techniques such as profilometry and neutron radiography. Clad and fuel microstructure and clad mechanical properties were investigated with destructive PIE techniques including ceramography, metallography, microhardness and ring tensile testing. It was observed that whilst dry-out had not affected the fuel microstructure, significant changes had been induced in the clad. These included high temperature corrosion resulting in moderate growth of the outer surface oxide layer and H_2 pick-up (hydriding formation). Some of the rods also exhibited uniform and localised clad creep-down into pellet-pellet interfaces and in the most severely tested rods that clad had undergone the α to β phase transformation. This material exhibited reduced UTS and brittle fracture. However, significant improvements of ductility were observed in clad that had been exposed to less severe in-pile transients where a small α phase grain structure was retained and hydrogen pick-up was minimal. None of the rods failed, neither during the dry-out phase or the following steady-state normal operation.

The test series has been completed. Similar test related to PWR conditions are being considered.

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4.2.2 Ramp Testing of Irradiated Fuel with Hydrided Cladding, (Sweden)

The objectives of this work are to study cladding integrity under PCMI loads in normal operation and anticipated transients, in particular to determine the effect of cladding hydriding on the cladding integrity (ramp failure threshold).

Research is carried out by the Studsvik Nuclear and sponsored by the Swedish Nuclear Industry Research Programme/Fuel Group (BFUK/PGB).

Ramp tests have been performed on PWR and BWR fuel rods with elevated levels of cladding hydriding. Complementary mechanical testing has been performed and is still ongoing. Two separate mechanisms of hydrogen-assisted cracking of fuel cladding under PCMI loads have been identified. These are brittle fracture of hydrides in heavily hydrided areas (with tearing of non-hydrided metal ligaments between the hydrides), and a time-dependent delayed hydride cracking (DHC) mechanism for crack propagation in less heavily hydrided material. In the future complementary mechanical testing is planned to be performed in order to determine the hoop strain required for crack initiation as a function of local hydrogen concentration at the cladding surface.

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5. LOCA Studies

5.1 Integral Tests

5.1.1 Blow-down Test Facility (BTF) Experiments, (Canada)

The BTF experiments were performed in the NRU reactor at the Chalk River Laboratories of AECL. The CANDU Owners Group (COG) sponsored the BTF experiments under joint funding from Ontario Power Generation, Hydro Québec, New Brunswick Power, and AECL.

Four BTF in-reactor experiments were performed to improve our understanding of CANDU fuel and fission-product behaviour under accident conditions, and to provide data for use in reactor safety code validation [1]. All four BTF tests have been completed, and final analysis and documentation is in progress.

BTF tests are conducted in the NRU reactor at the AECL Chalk River Laboratories [1]. The insulated test section is oriented vertically in a Zircaloy re-entry flow tube, which fits inside a thick-walled stainless-steel pressure tube located in the reactor core. The BTF test section and the NRU reactor core have separate heat transport systems. Test assemblies in the BTF are cooled with pressurized water or saturated steam. An accident sequence is initiated by isolating the in-reactor test section from the rest of the loop, and voiding the coolant through an instrumented blow-down line and a wire-mesh filter into a sealed tank in the basement of the reactor building. Steam, inert gas and cold water may be used for post-blow-down cooling in the BTF. The blow-down line is instrumented to measure coolant thermal-hydraulic parameters and fission-product gamma emissions.

In the BTF-107 experiment performed in 1990, a three-element cluster of CANDU-sized fuel elements was subjected to severely degraded cooling conditions resulting in a high-temperature (=2770 K) transient [2,3]. A partial flow blockage developed during the test due to relocation of a molten U-Zr-O alloy and the high-temperature transient was terminated with a cold-water quench. The results of the BTF-107 test were used in development and validation of our safety thermal hydraulics and fuel behaviour codes [3,4].

The other three experiments in the BTF program, BTF-104, BTF-105A and BTF-105B, were conducted with single CANDU-sized fuel elements at maximum temperatures of 1800-2200 K in a steam-rich environment. The tests were performed to evaluate the behaviour of a CANDU fuel element and the resultant fission-product release and transport in a LOCA/LOECC scenario.

The BTF-104 experiment performed in 1993, provided data on fuel behaviour, and volatile fission-product release and transport (Kr, Xe, I, Cs, Te and Ba) from a previously irradiated fuel element at a volume-averaged fuel temperature of about 1800 K [5-8].

The BTF-105A experiment was conducted in 1996, using an internally instrumented fresh fuel element [9-12]. The BTF-105A test provided data for validation of transient fuel performance codes and tested instrumentation for the BTF-105B experiment. The BTF-105A test data were used by Ontario Power Generation to validate the FACTAR fuel and fuel channel safety analysis code [13], and will also be used for validation of the ELOCA-IST transient fuel behaviour code.

The BTF-105B experiment was performed in 1997 to investigate fission-product release and transport from a previously irradiated fuel element at an average fuel temperature of 2100 K [14,15]. Due to improved measurements of fuel-cladding temperature, flow and neutron flux, and better control of steam condensation in the test section, the thermal hydraulic boundary conditions for the BTF-105B test are better quantified than for previous BTF tests.

The fission-product release, deposition and aerosol measurements from the BTF-104 and BTF-105B tests are being used to validate the SOURCE and SOPHAEROS codes. The BTF pressure tube was removed from the NRU reactor in 2000 June, marking the end of the BTF experimental program.
The final analysis and documentation of the BTF tests is scheduled for completion in 2001.

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5.1.2 Ballooning Test for Investigation of the ECC Coolability Criterion for Zr1%Nb Cladding, (Hungary)

In the absence of information on the ballooning behaviour of the Zr1%Nb cladding, KFKI Atomic Energy Research Institute of Hungary, sponsored by the National Committee for Technological Development, carried out a programme to clarify whether, in a large break LOCA, the coolability of a VVER core is preserved.

A series of tests were executed on VVER type 7-rod bundle test sections. The tests revealed that the blockage of the flow area never exceeds 80% and from thermo hydraulic investigations it was known that 20% of cross section available for cooling is sufficient to reflood the reactor core.

The programme has now been completed and no further action is planned.

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[1] L. Maróti: Quench test for the investigation of the 17% oxidation criterion of postulated accidents. 6th Int. Quench Workshop, Karlsruhe, 10-12 October 2000.

[2] P.Windberg, I.Nagy, Z.Hózer, M.Horváth: Ballooning experiments with VVER bundle. Proceedings of Nucl. Energy in Central Europe, Bled, Slovenia, 11-14 September 2000.

5.2 Separate Effects Experiments

5.2.1 CEA TAGCIS, TAGCIR, HYDRAZIR and CINOG Programme Test Series, (France)

This series of LOCA separate effects tests carried out by the CEA was started in 1991 and continued to 2001. A brief description of each programme is given below. 'all the tests involved well defined double sided oxidation and thermal shock tests on empty 17 x 17 Zr-4 cladding samples in a steam environment.

The TAGCIS test series (1991-1993) included more than 110 thermal shock tests on as-fabricated or pre-oxidized Zr-4 cladding samples. This series investigated the thermal shock behaviour of un-irradiated cladding having an initial corrosion similar to the end of life corrosion of a high burn-up fuel rod.

The test series carried out between 1993 and 1996 included 25 oxidation and thermal shock tests on irradiated Zr-4 cladding samples from high burn-up fuel rods irradiated for 5 cycles up to 60-63 MWd/kg in a commercial EDF nuclear power reactor where the end-of-life waterside oxide thickness was in the range 60 to 100-120 μm.

The HYDRAZIR tests carried out between 1996 and 1999 included oxidation and thermal shock tests at different cooling rates on un-irradiated as-fabricated and pre-hydrided Zr-4 cladding samples containing 0 to 5000 ppm H_2. One of the objectives of this series was to investigate the impact of pre-accident hydriding on transient oxidation and quenching behaviour. The interest in the impact of hydrogen came from the TAGCIS tests that showed that the behaviour of irradiated Zircaloy under LOCA conditions could possibly be related to the hydrogen.

The most recent CINOG test series carried out 1997 to 2001 included oxidation and thermal shock tests on as-fabricated Zr-4 samples and M5 samples. This series is devoted mainly to the licensing of new or advanced alloys under LOCA conditions.

The experimental process for assessing the ECR value for the TAGCIS tests was based on the metallographic measurements of the different metallurgical phases layer thickness and the subsequent calculation of the total amount of oxygen present in the cladding. In the more recent HYDRAZIR and CINOG tests, the same test facility provided, in terms of the measured local cladding temperatures, both the high temperature oxidation kinetics through weight gain measurements and the final quenching behaviour.

In all cases, the cladding was assumed to have failed as soon as it could not withstand an internal over-pressure of 1 bar of argon, and small bubbles were detected on the sample surface. This methodology has been viewed as extremely conservative since such a threshold is well below the fragmentation limit.

5.2.2 Quench Tests for the Investigation of the ECC 17% Oxidation Criterion, (Hungary)

This work was performed by the KFKI Atomic Energy Research Institute of Hungary, sponsored by the Hungarian Atomic Energy Authority.

Ring compression tests revealed earlier embrittlement of the VVER type cladding material (Zr1%Nb) in comparison with the Zry-4. The question arose whether the 17% oxidation criterion accepted by the Hungarian authority is applicable for the Zr1%Nb cladding.

The test matrix covered the temperature range between 1000 and 1300°C, while the time interval of steam oxidation was between 300 and 18000 s. The experimental results proved that the 17% oxidation criterion represents a safe limit considering the real physical margin. It was also recommended that the Russian limit of 18% was also acceptable.

The series of tests has been completed, but it is expected that application of new materials like the Russian E635 will necessitate new tests.

References

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[2] Z.Hózer, M.Horváth, L.Maróti, L.Matus, I.Nagy, A.Pintér, P.Windberg: High temperature experiments with fuel materials. Nuclear Science and Technology in Hungary, October 2000, pp 89-93.

5.2.3 Ring Compression Tests for the Steam Oxidation Caused Embrittlement of Zr1%Nb Cladding Material, (Hungary)

This work was performed by the KFKI Atomic Energy Research Institute of Hungary, sponsored by the Hungarian Atomic Energy Authority.

The aim of the research activity was the determination of the degree of embrittlement caused by steam oxidation at high temperatures. For comparison Zry-4 cladding samples were also studied. The results showed higher embrittlement of the Zr1%Nb tubes at the same degree of steam oxidation. Complete embrittlement of the Zr1%Nb samples was reached at about 7% ECR while the same results for Zry-4 were obtained at more than 20% ECR. The temperature range of the steam oxidation was between 900 and 1200°C. Subsequent measurement of the H_2 uptake revealed higher absorption by the Zr1%Nb.

The work is now complete with no plans for future studies.

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[2] L.Maróti, L.Matus, P.Windberg, Á.Griger: Ambient and high temperature mechanical properties of Zr1%Nb cladding with different oxygen and hydrogen content. HPR-351/35, Loen, 1999.

5.2.4 Single Tube Ballooning Tests for the Determination of the Burst Pressure/Stress of Zr1%Nb Cladding, (Hungary)

This work was performed by the KFKI Atomic Energy Research Institute of Hungary, sponsored by the Hungarian Atomic Energy Authority.

The validation of the transient fuel behaviour code (FRAP-T6) required experimental results on the burst parameters of the VVER type cladding tube in a wide range of accidental conditions. To cover all the expected situations detailed investigation of the effect of the oxide layer thickness had to be measured.

Up to the burn-up values accepted at present, the measurement results are sufficient. If the burn-up is significantly increased then probably further information will be required to consider the enhanced internal corrosion of the cladding.

The single tube ballooning experiments resulted in a correlation for the burst stress as a function of temperature. The correlation was implemented in to the FRAP-T6 code. The improved code version presented very good predictive capabilities.

References

[1] Cs. Gyori: Development and application of FRAP-T6 VVER version. Int. Conference "Nuclear Energy in Central Europe 2000", Bled, Slovenia, 11-14 September 2000.

[2] Cs.Gyori, Z.Hózer, L.Maróti, L.Matus: VVER Ballooning Experiments. HPR-349/40

5.2.5 Research on High Burnup Fuel Behavior under LOCA Condition, (Japan)

This work is carried out in the Fuel Safety Research Laboratory of JAERI, (Japan Atomic Energy Research Institute) and is sponsored by MEXT, (Ministry of Education, Culture, Sports, Science and Technology).

The objective is to evaluate the influence of burn-up extension to a range exceeding the current limit of 55GWd/t for BWRs and 48GWd/t for PWRs on fuel behaviour under LOCA condition and to obtain a wide-range data base available for regulatory judgment. The integral thermal shock test aims to investigate the cladding resistance to failure during a LOCA in order to discuss the applicability of current ECCS criteria, 15% ECR.

The program consists of oxidation rate measurement tests, tube burst tests, cladding mechanical properties tests and integral thermal shock tests. The comprehensive series of experiments use, pre-oxidized, pre-hydrided and/or irradiated claddings samples. The integral thermal shock tests simulate the whole LOCA sequence including cladding-burst, double-sided cladding oxidation and thermal shock by reflooding. In the tests, a test rod is quenched under fully or partially restraint condition to simulate possible tensile loading on a fuel rod due to its shrinkage in a grid span.
The oxidation tests, the mechanical properties tests, and the integral thermal shock tests with the pre-hydrided cladding (non-irradiated) have been performed to investigate separate effects of hydrogen absorption during normal operation. Preparation for irradiated cladding is progressing.

Results obtained so far are as follows:

- The influence of pre-hydriding on the oxidation rate varies depending on the oxidation temperature and the hydrogen concentration. The largest influence is observed at 1173 and 1223 K and the enhancement by pre-hydriding is estimated to be less than 5% for the hydrogen concentration lower than 800 wtppm in several minutes.
- Hydride redistribution and morphology change take place by temperature changes expected in a LOCA. This can greatly affect the mechanical property of cladding.
- ß-quenched cladding with high hydrogen concentrations exhibits very low ductility.
- As a result of the integral thermal shock tests varying hydrogen content and axial load during quenching, it was shown that;
  - The failure threshold generally decreased with the increase in axial tensile load.
  - The influence of pre-hydriding was obviously seen on the failure threshold value under restraint conditions.

The failure threshold value higher than 20% ECR was obtained for axial tensile load condition below 600N and for the hydrogen concentration range examined

Future tests will be performed with irradiated cladding tube. A computer code will be developed in a couple of years to analyze the failure behaviour of a fuel rod during a LOCA based on the results from the current research program and international co-operation. The experimental and analytical investigation will be summarized by 2004, though further work is envisaged to support higher burn-up, in the range up to 75 GWd/t and advanced designs of fuel.
The sponsor of this work is METI (Ministry of Economy, Trade and Industry).

References

[1] Nagase, F., Otomo, M., Tanimoto, and Uetsuka, H., “Experiments on High Burnup Fuel Behavior under LOCA Conditions at JAERI”, ANS topical meeting on LWR fuel performance, April 10-13, 2000, Park City.

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5.2.6 ANL Integral LOCA Tests for Limerick BWR Fuel Rods, (USA)

The primary purpose of these tests is to evaluate the performance of high burn-up fuel relative to the NRC cladding embrittlement criteria defined in 10CFR50.46:

(1) The calculated maximum fuel element cladding temperature shall not exceed 2200 F
(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before steam oxidation.

Within the test plan, the tests will be conducted on fuel rod segments (300 mm long) with the as-irradiated cladding outside diameter and cladding inside diameter oxide layers and the fuel intact. In this way the high burn-up effects of the oxide layers, the associated hydrogen pick-up due to waterside corrosion and the fuel cladding contact and/or bonding will be present in the tests. The central 150-200 mm of the test sample will be uniformly heated. The specimens will be pressurized, stabilized at 300 C, heated at 5 C/s to 1204 C, held at 1204 C at a time corresponding to a calculated equivalent cladding reacted (ECR) of 17%, slow cooled to 750 - 800 C and water quenched. The calculation of the ECR versus time at 1204 C will be made using the ANL A model, with model parameters adjusted based on the results of the oxidation test at 1204 C.

Three tests will be run, the first with ECR ~17%, the second with ECR ~30% and the final one at an intermediate level. Up to three additional tests may be run, depending on the results of the first series. During the 5 C/s rise to 1204 C, the cladding will balloon and bursts. Where practically possible, additional measurements will be made during this phase of the test, e.g., diameter and axial extent of the balloon, form and degree of fuel relocation and the effect of these phenomena on circumferential and axial temperature profile. Post test measurements include: the ECR and its comparison with the calculated value, hydrogen content and phase distribution.

5.2.7 HRP LOCA test series with high burnup PWR and BWR fuel, (Norway)

The OECD Halden Reactor Project has conducted several series of experiments related to coolant transients. Both fresh and irradiated fuel segments were employed in single rod and sub-bundle tests in the past /1/. A new LOCA test series utilising high burnup fuel segments is being prepared within the Joint Programme of the Halden Project. Based on the discussions and recommendations of two workshops /2, 3/, the objectives of the HRP LOCA experiment are as follows:

(1) Observe overall fuel behaviour under expected and bounding temperature conditions in an in-reactor test.
(2) Measure extent and timing of relocation of fuel fragments into the ballooned region and any consequences of such relocation.
(3) Measure the extent of "secondary transient hydriding" in the vicinity of the ballooned region.

Two temperature levels will be aimed at: 800 ^{o}C to assess realistic conditions (high burnup fuel is not expected to reach high temperature in a LOCA) and more bounding conditions of 1100 ^{o}C. The tests will first be run at the moderate temperature, 800 ^{o}C, to observe ballooning deformation without subsequent high-temperature effects.

Temperatures, heat-up rates, pressures, void volumes, and times will be matched to those in the ongoing ANL program to facilitate the identification of any phenomena that might not be adequately represented in out-of-reactor tests.

Both BWR and PWR high burnup fuel rods irradiated in commercial reactors to burnup levels >50 MWd/kg will be investigated. Together with potentially high rod internal pressure, fuel-clad bonding is one of the major differences compared to fresh or low burnup fuel. It is also foreseen to conduct an additional test with lower burnup fuel (about 40 MWd/kg, less or no fuel-clad bonding) in order to provide an intermediate point between low and high burnup.

The Halden testing focuses on in-reactor effects that are different from those obtained in out-of-reactor tests. In particular the heating from within may affect phenomena such as axial gas flow, fuel bonding and fuel relocation. The longer segments to be utilised in the in-reactor tests compared to planned hot cell programs could also have an impact on axial gas flow.

Test Rig Design and Instrumentation

With background on the IFA-54X series of LOCA experiments and the IFA-613 dry-out test series, the design of the LOCA test rig includes the following features:

- Single rod with high burnup fuel.
- Heating provided from within the rod by locally controlled nuclear power generation simulating decay heat.
- Simulation of the thermal boundary conditions and constraints provided by neighbour rods by utilising an insulating channel.
- Temperature control with slightly heated shroud and a spray system (in addition to power control).
- Instrumentation includes cladding surface thermocouples, a cladding extensometer to provide data on onset of local overheating and dimensional changes, and pressure sensor. More substantial axial fuel relocation in-pile can be detected by neutron detectors.

A trial test execution utilising fresh fuel is foreseen for Spring 2003 followed by runs with high burnup PWR (2) and BWR (2) fuel. Efforts are also extended to include VVER fuel segments in the test series at a later stage.

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6 RIA Studies

6.1 Integral Tests

6.1.1 The CABRI REP Na Programme, (France)

This programme was initiated in 1992 by the “Institut de Protection et de Sûreté Nucléaire”, (IPSN) [1] [2] [3], and was conducted in collaboration with Electricité de France (EDF) and with a participation of the US NRC.

The main objectives were to study the potential high burnup effects on UO_2 fuel behaviour, analyse the MOX fuel behaviour and verify the adequacy or modify the present safety criteria previously defined for lower burnup fuel.

As a first step, the CABRI-REP Na experimental programme was initiated in the sodium loop of the CABRI reactor [4] in which consequences of a fast power transient applied to a single rod could be studied in a sodium coolant environment. Due to this last point, the investigation was focused on the first phase of the power transient when strong pellet-clad mechanical interaction (PCMI) occurs with limited clad heat-up.

In parallel to the REP Na experiments, the SCANAIR code [5] was developed in order to interpret the test results, perform sensitivity studies and translate the results to reactor conditions. Separate-effects tests are also being conducted for the study of the cladding mechanical properties (PROMETRA [6]), the clad-fluid heat transfer under fast transients (PATRICIA) and the transient behaviour of fission gases (SILENE-RIA). These are all discussed in later sections.

From 1993 to 1998, seven tests with UO_2 fuel and three tests with MOX fuel were performed using mostly refabricated rods from PWR fuel, with low internal pressure (0.3 MPa He pressure, consistent with sodium channel pressure). The coolant conditions were : inlet temperature of 280°C, fluid velocity of 4 m/s and channel pressure of 3b.

The following parameters were studied :

- rod burn-up from 33 to 64 GWd/t,
- cladding corrosion thickness from 4 to 130 µm of ZrO_2,
- corrosion conditions from uniform to spalled with hydride blisters resulting from reactor operation,
- power transients starting from almost zero power to different energy levels and with various pulse widths leading to different energy injection rates.

The test matrix gathering the test characteristics and the main results is given in Table 6.1.1. The total set of measurements is composed of pre-test examinations, in-pile transient diagnostics and post-test examinations (gamma-scanning, x-ray, profilometry, eddy-current, gas puncturing and analysis, and destructive examinations).

In the year 2000, two additional tests using EDF rods were performed in June and November respectively :

- the REP Na 11 test using a UO_2-5 cycles rod (64 GWd/t local maximum) with M5 cladding and a maximum energy injection of 104 cal/g with a pulse width of 31 ms

- the REP Na 12 test using a MOX-5 cycles rod (64 GWd/t local maximum, MIMAS-AUC process) with a zircaloy-4 cladding (80µm corrosion thickness without spalling) and a maximum energy injection of 104 cal/g with a pulse width of 63 ms.

Both tests did not lead to evidence of rod failure and the detailed analysis of their results is underway.

As a first outcome, the REP Na tests which led to rod failure at fuel enthalpy levels (radial average) from 30 cal/g (REP Na 1) to 120 cal/g (REP Na 7) underlined the lack of adequacy of the present safety criteria for high burnup UO_2 and MOX fuel.

Similar conclusion was derived from tests performed in the NSRR facility in Japan and showing low enthalpy failure level of high burnup UO_2 fuel.

On the other hand, the detailed analysis and interpretation of the first REP Na tests allowed identification of the physical mechanisms and the key parameters influencing the fuel behaviour.

A deleterious effect of a high clad corrosion level with spalling and hydride concentration, reducing the clad ductility (also confirmed by the PROMETRA mechanical testing) has been revealed; on the other hand, moderate or large corrosion without spalling (up to 80 µm) may withstand significant clad straining which is obtained as a result of fuel thermal expansion and fission gas swelling and depends on the level of energy injection.

The contribution of fission gases on clad loading in addition to the classical fuel thermal expansion effect, has been underlined: such gas contribution and fission gas release are increased with burnup and in case of MOX fuel due to its non-homogeneous, structure with $UPuO_2$ agglomerates. Indeed, extensive fuel fragmentation (grains separation) has been observed in most of the REP Na tests. This phenomenon is attributed to the high overpressure which is developed in the small inter-granular bubbles during fast heating rates and which induces high stress fields between the grains, leading to the grain boundaries cracking. Subsequent grains separation depends on the respective influences of gas pressure and external fuel constraint. Largely observed in UO_2 fuel, it appears also clearly in the fuel matrix with MOX tests, in spite of the relatively low burnup level. The main consequences of this phenomenon are a degradation of fuel mechanical properties, the fast availability of all the grain boundary gases with associated driving pressure leading to solid fuel pressurization and swelling, clad loading with risk of failure and finally to gas release.

Due to the high gas content in inter-granular and porosity bubbles associated to agglomerates behaviour under irradiation, solid fuel pressurization and clad loading may be increased in MOX fuel and may explain the failure of REP Na 7 suggesting a high burn-up effect with MOX fuel.

The influence of the energy injection rate (pulse width) on the clad loading and on the potential for fuel dispersal in case of rod failure has been pointed out (REP Na1 – REP Na 8 and REP Na 10).

The possibility of transient clad oxide spalling linked to clad initial corrosion and clad straining has also been evidenced and may be of concern in water environment by the possible influence on boiling crisis occurrence.

6.1.2 The CABRI Water Loop Programme, (France)

Although the CABRI-REP Na program has been the basis for the EDF authorization to increase the fuel burnup up to 52 GWd/t (mean assembly), questions still remain about the effect of DNB on cladding failure, the influence of internal rod pressure, and the possibility of fuel-coolant interactions after failure.

The complexity of these phenomena and their important coupling make it difficult to have confidence in the current results without experimental confirmation with integral tests under representative PWR conditions. Moreover, lack of representativity was also identified in separate effect tests such as PROMETRA mechanical program under RIA conditions, preventing reliable prediction of the rod behaviour on the sole basis of such experiments at the present time. Prototypical PWR conditions are particularly important for the qualification of any further increases in fuel burnup in power reactors.

These are the reasons that IPSN (now IRSN) has decided to replace the present sodium loop in CABRI with a pressurized water loop (PWL) and to propose an international program called CABRI-International Programme (CIP).

Twelve tests have been proposed for the CABRI- International program and they include high-burn-up fuel tests combined with mechanical tests to provide the understanding necessary to extrapolate to a broad spectrum of reactor conditions. Six test series have been identified :

CIP 0 — two tests in the sodium loop using advanced fuels (rods with Zirlo and M5 cladding already committed)

CIP 1 — two tests in the water loop with the same advanced fuels to provide a link to CABRI- REP-Na

CIP 2 — tests with ultra high burn-up fuel (80-100 GWd/t, one DUPLEX rod committed)

CIP 3 — tests specifically designed to improve the understanding of RIA phenomena

CIP 4 — tests with MOX fuel

CIP 5 — complementary tests (open)

It is to be noticed that among these tests, the first test to be realized in the pressurized water loop will be considered as a qualification of the loop : it will be defined with a particular attention and will be performed with a rod having a Zircaloy-4 cladding.

These integral tests will be coupled with separate-effect tests (mechanical testing, fission gas behavior experiments) and code development to facilitate translation to power reactor conditions, development of safety criteria or limits and evaluation of the safety margins.

The CABRI facility will be renewed but, taking into account the urgent need from utilities and safety experts, the first tests of the S0 series will be realised in the sodium loop in 2002.

The other test series will be performed in 2005-2007 after renovation and installation of the Water Loop.

The CABRI PWL program is now settled as an international program and agreements are being signed (Umbrella agreement under the auspices of OECD and bilateral agreements for financial provisions).

The key feature of this program will be to provide testing of advanced fuels in the same conditions and in international context with both participation of industry and regulatory agencies.

References

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TABLE 6.1.1

The CABRI REP Na Tests - UO_2 Fuel

Test	Rod	Pulse (ms)	Energy end of peak (cal/g)	Corrosion (µ)	RIM (µ)	Results and observations
Na-1 (11/93)	GRA 5 4.5 % U 64 GWd/t	9.5	110 (at 0.4 s)	80 important initial spalling	200	- Failure, brittle type for H_f = 30 cal/g - Hydride accumulation - Fuel dispersion 6 g, including fuel fragments outside RIM (> 40 µ) - Pressure peaks in Na of 9-10 bars
Na-2 (6/94)	BR3 6.85 % U 33 GWd/t	9.1	211 (at 0.4 s)	4		No failure Hmax = 210 cal/g Max. strain : 3.5 % average,3.1% mid pellet FGR : 5.5 %
Na-3 (10/94)	GRA 5 4.5 % 53 GWd/t	9.5	120 (at 0.4 s)	40	100	No failure Hmax = 125 cal/g Max. strain : 2 % FGR : 13.7 %
Na-4 (7/95)	GRA 5 4.5 % U 62 GWd/t	# 75	95 (at 1.2 s)	80 no initial spalling	200	No failure Hmax = 99 cal/g Cladding spalling under transient Max. strain : 0.4 % FGR : 8.3 %
Na-5 (5/95)	GRA 5 4.5 % U 64 GWd/t	9.5	105 (at 0.4 s)	20	200	No failure Hmax = 115 cal/g Max. strain : 1.1 % FGR : 15.1 %
Na-8 (07/97)	GRA 5 4.5 % U 60 GWd/t	75	106 (at 1.2 s)	130 lim. initial spalling	200	Failure $H_f \leq$ 82 cal/g , Hmax=110 cal/g no fuel dispersion
Na-10 (07/98)	GRA 5 4.5 % U 62 GWd/t	31	107 (at 1.2s)	80 important initial spalling	200	Failure at H_f = 79 cal/g, Hmax = 110 cal/g No fuel dispersal - Examinations still underway
The CABRI REP Na Tests - MOX Fuel						
Na-6 (03/96)	MOX 3 cycles 47 GWd/t	35	126 à 0.66 s 165 at 1.2 s	40		No failure Hmax = 148 cal/g Max. Strain : 2.65 % FGR : 21.6 %
Na-7 (1/97)	MOX 4 cycles 55 GWd/t	40	125 à 0.48 s 175 à 1.2 s	50		Failure, H_f= 120 cal/g Strong flow ejection, pressure peaks of 200-110b, fuel motion in the lower half zone;
Na-9 (04/97)	MOX 2 cycles 28 GWd/t	34	211 at 0.62 s 241 at 1.2 s	< 20		No failure Hmax = 210 cal/g Max.strain : 7.3% (mean) FGR : 35%

6.1.3 RIA Experiments in the NSRR (Nuclear Safety Research Reactor), (Japan)

This programme is carried out in the Fuel Safety Research Laboratory in the Department of Reactor Safety Research of JAERI (Japan Atomic Energy Research Institute) sponsored by MEXT.

The objectives of this program are:

- to investigate fuel behaviour of irradiated fuel rod under RIA condition,
- to determine the fuel rod failure threshold of irradiated fuel rod and to clarify the influences of the fuel burn-up, and
- to clarify the mode, mechanism and consequence of the failure of irradiated fuel rod.

The program has provided a database for the current regulatory guide for reactivity-initiated events in light water reactors.

In twenty-three experiments with irradiated PWR fuels, seven different test specimens have been re-fabricated from full-size commercial reactor fuels, and subjected to a pulse irradiation. The test fuels consist of the fuels irradiated in the Mihama (MH), Genkai (GK) and Ohi (OI) reactors as well as High Burn-up fuels irradiated in the Ohi reactor (HBO) and Takahama reactor (TK) fuels. The HBO and TK fuels include types A and B fuels that are manufactured by different fuel vendors. Burn-ups of the PWR test fuels range from 38 to 50 MWd/kgU. The TK test fuels have low-tin, 1.3%Sn, Zry-4 cladding, and others are with 1.5%Sn Zry-4.

Fourteen experiments with irradiated BWR fuels comprised of two test series, i.e. five Tsuruga (TS) experiments for 7x7 type 26 MWd/kgU rods and nine Fukushima (FK) experiments for 8x8 type rods. In the first three tests of the FK test series, tests FK-1 FK-2 and FK-5, were conducted with 8x8BJ Step I type rods with Zr-liner cladding at burnups of 41 to 45 MWd/kgU irradiated for 5 cycles. The subsequent two tests, FK-4 and FK-5, used 8x8 Step II type rods with Zr-liner cladding at a burnup of 56 MWd/kgU irradiated for 4 cycles. The recent four BWR fuel experiments, tests FK-6 through FK-9, were performed with 8x8 Step II type rods with Zr-liner cladding at a burnup of 61 MWd/kgU irradiated for 5 cycles.

As for MOX fuels, four experiments with 20 MWd/kgHM fuels irradiated in ATR ‘Fugen’ were conducted.

In addition, short fuel rods pre-irradiated in the Japan Materials Testing Reactor (JMTR) of JAERI were also subjected to the pulse irradiation in 22 experiments of JM, JMH and JMN test series.

An overview of the tests performed is given in Table 6.1.2, taken from presentations at the ANS Meeting, Park City, April 2000.

The results obtained to date can be summarized as follows:

- The JM-4 and JM-5 experiments in FY1990 showed that cladding failure of irradiated fuels occurs due to PCMI (Pellet-Cladding Mechanical Interaction). The fact triggered French sodium loop test, Cabri REP Na program.

- The HBO-1 with 50 MWd/kgU fuel performed 4 months after the Cabri REP Na-1 showed hydride assisted PCMI failure in high burnup PWR fuels. The results of the both experiments appear to indicate the reduction of failure threshold at high burnup.
- In the HBO and TK tests with high burnup PWR fuel rod segments, test specimens with thicker oxide layer from the higher elevation, failed at the values as low as 251 J/g (60 cal/g) of fuel enthalpy. The results indicate a key role of fission gas loading, fission-gas-induced expansion due to high pressure loading of heated fission gas confined in grain boundaries.
- In the experiments resulting in fuel failure, fuel fragmentation and mechanical energy generation were observed. Only in the NSRR experiments, generated mechanical energy and particle size of fuel debris are measured and quantitatively evaluated.
- The recent BWR fuel experiments resulted in significant cladding failure and fuel dispersion. Although the data are preliminary, the results suggest that the occurrence of intensive PCMI loading is due to the bonding. Radially located hydride clusters may have an influence on the cracking of the cladding.
- The MOX fuel tests showed larger rod deformation and fission gas release, probably due to Pu-rich clusters in MOX fuels.

Future plans for the programme include:

- Higher burnup fuel specimens are to be subjected to the NSRR experiments which include PWR fuels up to 74 MWd/kgU with ZIRLO, MDA and Low-tin Zry-4 cladding, BWR fuels up to ~75 MWd/kgU with Zry-2 cladding with Zr-liner, PWR MOX fuels up to 59 MWd/kgHM, and BWR MOX fuels up to 76 MWd/kgHM. These fuels will be transferred from European LTA program in FY2003. This series of tests is named as Advanced LWR fuel Performance and Safety research (ALPS) program and is sponsored by METI.
- Japanese fuels from 55MWd/kgU LUA PWR and BWR assembles will be tested in the NSRR.
- 30 to 40 MWd/kgHM ATR/MOX fuels will also be subjected to the pulse irradiation.
- The capsule for high-temperature and high-pressure is under development, and its licensing will start from FY2002

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Table 6.1.2(a) NSRR PWR tests as reported at ANS Park City Meeting April 2000

Test	Burnup (GWd/tU)	Oxide Layer (microns)	Fuel Type	Pulse Width (msec)	Max. ΔH (cal/gm)	ΔH at Failure (cal/gm)	Fuel Dispersal
HBO-1*	50	40-50	17x17	4.4	73	60	Yes
HBO-2*	50	30-40	17x17	6.9	37		
HBO-3*	50	22	17x17	4.4	74		
HBO-4*	50	18	17x17	5.4	50		
HBO-5*	44	35-60	17x17	4.4	80	77	Yes
HBO-6*	49	20-30	17x17	4.4	88		
HBO-7	49	30-50	17x17	4.4	88		
MH-1	39	5	14x14	5.3	47		
MH-2	39	5	14x14	5.0	55		
MH-3	39	5	14x14	4.8	67		
GK-1	42	10	14x14	4.8	93		
GK-2	42	10	14x14	4.8	90		
OI-1	39	N/A	17x17	4.4	106		
OI-2	39	N/A	17x17	4.4	108		
TK-1*	38	7	17x17	4.4	125		
TK-2	48	15-35	17x17	4.4	107	60	Yes
TK-3	50	8	17x17	4.4	99		
TK-4	50	20	17x17	4.4	98		
TK-5	48	25	17x17	4.4	101		
TK-6	38	15	17x17	4.4	125		

* - Analytical evaluations were performed for these tests.

Table 6.1.2(b) NSRR BWR tests as reported at ANS Park City Meeting April 2000

Test	Burnup (GWd/tU)	Oxide Layer (microns)	Fuel Type	Pulse Width (msec)	Max. ΔH (cal/gm)
TS-1	26	6	7x7	6.7	55
TS-2	26	6	7x7	6.2	66
TS-3	26	6	7x7	5.6	88
TS-4	26	6	7x7	5.0	84
TS-5	26	6	7x7	4.5	98
FK-1*	45	20-40	8x8BJ	4.4	130
FK-2*	45	20-40	8x8BJ	5.3	70
FK-3	41	20-40	8x8BJ	4.4	145
FK-4	56	20-40	8x8	4.4	140
FK-5	56	20-40	8x8	5.3	70

*- Analytical evaluations were performed for these tests.

6.2 Separate Effects Studies: Fuel

6.2.1 IRSN SILENE-RIA Programme, (France)

The SILENE-RIA programme defined by IRSN, is being carried out in the CEA facility at Valduc and is a complementary programme to the CABRI experiments. It is devoted to the study of fission gas transient behaviour .

The analysis of the CABRI-REP Na tests has underlined the role of fission gases on the high burnup behaviour under RIA. In particular, a strong influence of the grain boundary gases on the clad loading, in addition to PCMI loading, is clearly suggested in the rim zone of a high burnup UO_2 fuel and from the $UPuO_2$ clusters of MOX fuel.

However, insufficient knowledge is presently available to quantify this potential loading and separate effect experiments have been defined in order to obtain quantitative information on :

- fuel fragmentation and associated loading mechanisms with estimation of driving pressure,
- gas release kinetics and identification of the main parameters.

These tests will be performed in the SILENE reactor using a double-wall capsule with two independent cells and various on-line instrumentation: thermocouples, pressure transducers, acoustic and strain sensors. Pre and post tests measurements will be also performed. Different capsule designs and fuel-clad geometry are foreseen which should allow to reach the various objective :

- thin slices with expansion volume to study the fuel fragmentation and the associated fuel expansion or dispersion,
- thin slices without expansion volume to quantify the driving force from fission gases,
- fuel pieces (10 cm height) to determine the fission gas release kinetics, using intact rod piece or modified geometry specially designed for analysis of rim behaviour, radial and axial transfers under representative restraint conditions and study of the influence of fuel microstructure change on gas flow.

Starting from room temperature, the “pulse” operation mode in SILENE reactor leads to a rapid power excursion (width ~ 6 ms), and a fast energy injection in the tested fuel. Fuel-clad cooling occurs by heat transfer across solid body and gaseous gaps. Major on-line diagnostic comes from the pressure transducer signal, which, depending on the capsule and fuel-clad design, will give an indication on the gas release kinetics or on the fuel dispersion and expansion rate.

Presently, the test matrix includes twenty tests, using high burn-up UO_2 fuel (5 cycles) and MOX fuel (3 and 4 cycles, coming from the father rods of REP-Na6 and Na7). Two filling pressure conditions (0.1 and 5 MPa) and the different capsule and fuel-clad designs would allow to obtain a better understanding of the high burn-up UO_2 fuel and MOX fuel under RIA transients, in spite of the relatively low performance of the SILENE reactor in terms of energy deposition. The first tests with irradiated fuel are planned during second semester 2001.

6.2.2 Separate Effects Tests on Fuel in the NSRR Facility, (Japan)

Within the scope of the RIA test Programme in the NSRR facility, there are a series of separate effects tests being conducted in addition to the whole rod irradiations.

Considerable fission gas releases and large hoop deformation were observed in pulse irradiation tests of high burnup fuels at the NSRR under simulated RIA conditions. Significant grain boundary separation was seen in the post-test fuels with the large deformation. Thus, fission gases accumulated at the grain boundaries during the base irradiation are believed to be the primary sources for the deformation and the gas releases. However, thresholds for the grain boundary separation and for the gas release are not known. A set of separate effect tests to investigate the threshold under various constraint conditions by cladding are being prepared. Round slices of high burnup fuel will be pulse-irradiated in the NSRR. Pellet Transient pressure change due to the fission gas release and the post test fuel morphology at various enthalpies will be examined in the tests.

Particle fuel experiments were carried out in the NSRR to demonstrate mechanical energy generation due to thermal interaction between solid fuel fragments and coolant, and to clarify dependence of thermal to mechanical energy conversion ratio on fuel particle size. Non-irradiated UO_2 particles of 30 g were packed in a vinyl bag with water and subjected to pulse irradiation. Average particle size was varied from 20 to 250 µm. The mechanical energy generated was measured as the maximum kinetic energy of the jumping water column in the test capsule. Results from four experiments with different particle sizes clearly showed the dependence; the finer particles caused the higher energy conversion ratio. The highest conversion ratio obtained is 0.41% for the particles with average diameter of 20 µm. Extrapolation for these results suggests the conversion ratio of approximately 1% for 10-µm particles, which is the initial size of fuel grain.

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6.3 Separate Effects Studies: Cladding

6.3.1 IRSN PROMETRA and PATRICIA Test Programmes, (France)

The PROMETRA test programme aims at the determination of the clad mechanical properties of Zr-4 standard under RIA conditions, for implementation into the SCANAIR code. The stress-strain laws and a failure criterion are expected from hoop and axial tensile tests on two wing samples from defueled cladding and spark machined in the tranverse and rolling directions. The main outcomes of the programmes have been presented at the SMIRT 15 [1].

The test parameters were: the corrosion thickness (0 to 130µm) and the state of corrosion (with and without initial spalling), the strain rate ranging from 0.01 to $5s^{-1}$ and the temperature (280°C to 600°C (hoop) up to 1100°C for axial test, some points at 20°C and 150°C).
The results mainly showed that clad ductility is drastically reduced in case of very high corrosion and oxide spalling and the brittleness is linked to hydride concentration (blisters).
In case of absence of spalling, the influence of corrosion thickness (up to 80µm) is small and the strain rate has a low effect on the ultimate tensile strength and on yield strength.

At the present time new samples are being designed in order to improve the failure data and will be used for the future testing of the advanced claddings (Zirlo, M5, …) in the frame of the CABRI-Water Loop programme.

The PATRICIA programme aims at the determination of clad to coolant heat transfer under fast temperature transients for implementation in the SCANAIR code. It has been performed in an out of pile facility using electrically heated rods of 60 cm length cooled by water flow. The power transient is produced inside an inconel cladding and thermal-hydraulic conditions reproduce PWR or low pressure conditions. Several tests with various kinetics of clad temperature transients (simulating those resulting from fuel to clad heat flux) have been performed. From the measured inner clad temperature, the coolant temperature and applying an inverse conduction method, the clad to coolant heat flux is deduced. The interpretation of the test is underway and will be presented in the near future.

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6.3.2 NSRR Separate Effects Studies: Effect of Cladding Preoxidation on Rod Coolability, (Japan)

These tests were conducted within the scope of the RIA test Programme in the NSRR facility.

A series of NSRR experiments with non-irradiated fuel rods were performed to evaluate the effect of a cladding oxide layer on rod coolability during an RIA. NSRR experiments with irradiated fuel rods showed cladding surface temperature lower than those observed in fresh fuel tests. A possible speculation for the temperature difference is that oxide layer at the cladding outer surface of irradiated fuel rods enhanced heat transfer at the surface. In order to verify the speculation, pulse irradiation tests were performed on three kinds of fuel rods with three different surface states; non-oxidized, with oxide layer of a 1-µm thickness, and with that of a 10-µm thickness. Transient records of the cladding surface temperature showed raised critical heat flux and raised minimum heat flux for the oxidized cladding. These effects depend on the presence of the oxide layer, not on the thickness of the layer. The results support the theory that the most possible mechanism of the enhanced heat transfer is wettability increase at the cladding surface due to oxidation.

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6.3.3 Cladding Mechanical Properties Testing in the USA, (USA)

Argonne National Laboratory (ANL) and The Pennsylvania State University (PSU) are working together on a NRC-funded program to investigate cladding properties and to test loss-of-coolant accident (LOCA) acceptance criteria at high burn-ups. Although the main focus of the program is to investigate fuel behaviour under LOCA conditions, related mechanical properties testing is being done under both LOCA conditions and rod ejection accident conditions. The tests at relatively low temperatures and high strain rates appropriate for rod ejection accident conditions are described briefly here.

The objectives are two-fold: to understand the degradation in cladding failure behaviour at high burnup and to obtain stress-strain relationships that will serve as inputs to codes. High-burnup fuel rods of about 70 GWd/t from the H. B. Robinson PWR are expected to be available for these tests along with related archive fresh tubing. Although the fuel has not arrived at the time of this writing, high-burn-up specimens (about 50 GWd/t) from TMI-1 are available and have been used for preliminary testing along with non-irradiated Zircaloy-4 tubing.

*Ring-Stretch Tests***.** A ring tensile specimen design has been developed and tested at ANL to generate tensile properties in the hoop direction. A related ring specimen design was developed and tested at PSU to provide a near plane-strain stress state that approximates the stress state produced by expanding fuel pellets during an RIA. Tensile testing of cladding samples from archival tubing and high burn-up rods will be performed over a temperature range from room temperature to 800^{o}C with strain rates from 0.1%/s to 100%/s on irradiated an non-irradiated specimens. Because hydrogen is expected to play an important role on the mechanical properties of the irradiated material, testing is also being performed by PSU on artificially hydrided specimens of non-irradiated materials. These artificially hydrided samples allow an investigation not only of hydrogen content, but hydrogen distribution, i.e., when concentrated in a hydride rim or in blisters. Stress-strain relationships, along with tensile strengths (yield and ultimate) and elongations (uniform, total, and local) will be measured as a function of temperature, strain rate, radiation damage, hydrogen, and oxygen content.

Axial Tensile Tests. Similar testing will be performed on axial tensile specimens electro-machined from de-fuelled portions of irradiated fuel rods and from non-irradiated tubing specimens. These tests will be performed over the same temperature range and strain-rate range as the ring-stretch tests. The combination of the axial and the hoop stress-strain properties will allow validation and improvement of the models used in fuel rod codes for predicting the mechanical behaviour of an anisotropic alloy such as Zircaloy.

Biaxial Tube Burst Tests. Biaxial tube burst tests are the most informative and the most difficult to perform, and they consume the largest amount of specimen material, which is a significant consideration when testing irradiated fuel material. These tests will be performed in a more limited 300°C–400°C temperature range, but they will explore the effects on deformation and failure of stress biaxiality ratios from 1:1 to 2:1 at high strain rate. In principle, the tests can be run with the fuel intact or with the fuel removed. Some tests will be run with the fuel removed to generate baseline data for code validation along with data that can be compared to other such studies on non-irradiated and medium-burnup cladding.

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6.3.4 Cladding Integrity Under PCMI Loading During RIA, (Sweden)

The objective of this project is to study the cladding fracture under simulated RIA PCMI loading as a function of temperature, cladding hydrogen concentration, and strain rate using the EDC ("Expansion Due to Compression") soft mandrel testing technique. The research is carried out by the Studsvik Nuclear and sponsored by the Swedish Nuclear Industry Research Programme/Fuel Group (BFUK/PGB) and by the Westinghouse Atom.

Irradiated PWR cladding specimens (Zircaloy-4 and Zirlo) with different local levels of corrosion and hydrogen concentration have been tested at room temperature and at 340 °C. Irradiated BWR cladding specimens with different local levels of corrosion and hydrogen concentration have been tested at temperatures from 20 °C to 300 °C and at different strain rates. For both the PWR and the BWR specimens tested the temperature and hydrogen contents were found to have a significant effect on the strain to failure. A methodology for determining the strain energy density (SED) from the tests has been developed. Further EDC tests on cladding specimens from high burnup BWR cladding will be performed in order to elucidate the effect of temperature at intermediate temperature levels.

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7. Fuel Performance Codes, Steady State and Transient Application

7.1.1 BNFL ENIGMA Development Programme, (UK)

The BNFL fuel development programme described in section 2.2 also includes the maintenance and development of the ENIGMA fuel performance code. ENIGMA has been in use for UO2 fuels for both water and gas reactor systems for more than a decade, and has been accepted by regulatory authorities for fuel licensing work in the UK and elsewhere. A re-assessment of the code for MOX modelling was initiated at the outset of the MOX development programme. Specific developments have been undertaken in fuel creep modelling, in the tracking of higher plutonium and other isotopes, in providing MOX-specific fission product spectra and in the modelling of helium generation and release. A major programme of work is currently being undertaken to develop an improved mechanistic description of fission gas behaviour. In-pile testing and post-irradiation annealing, in each case followed by detailed SEM and TEM studies, have helped to reveal the role played by both intra-granular and grain face gas bubbles in determining fission product release and swelling. Extensive studies carried out on UO2 are in the process of being repeated on MOX fuel.

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7.1.2 Canadian Fuel Safety Codes, (Canada)

Ontario Power Generation, Hydro Québec, New Brunswick Power, and AECL jointly participate in the development and validation of Canadian fuel safety codes. The CANDU Owners Group (COG) sponsors some of the fuel safety code development and validation. Other work is funded directly by AECL and the utility participants.

A systematic validation of the computer codes used in Canada for reactor safety analysis is now in progress [1]. The Industry Standard Toolset (IST) initiative has identified a standard set of computer codes for use in Canadian reactor safety analysis to allow the participating organizations to focus and coordinate their efforts on code development and validation [2]. The development and validation of Canadian fuel safety codes is performed to provide tools for simulating fuel behaviour and fission-product release under accident conditions for use in CANDU reactor safety analysis. The IST initiative is led by a steering committee consisting of senior representatives from AECL and the Canadian nuclear utilities. Two working groups are responsible for fuel-related code development and validation: fuel and fuel-channel thermal-mechanical behaviour, and fission-product release and transport.

Three main codes have been developed for fuel safety analysis in Canada: ELESTRES for simulation of fuel behaviour under normal operating conditions, ELOCA for simulating fuel behaviour under accident conditions, and SOURCE for simulating fission-product release from the fuel under both normal operating and accident conditions.

The main purpose of the ELESTRES code is to calculate temperatures, fission-gas release, and hoop strains in a fuel element during normal operating conditions [3,4]. The code models thermal, microstructural, and mechanical behaviour of nuclear fuel elements of the 28, 37 and 43-element bundle designs. The original version used a one-dimensional (radial) finite-difference model for thermal simulations, a point model for microstructural simulations, and a two-dimensional (radial-axial) finite-element model for mechanical simulations. For normal-burnup CANDU fuel, ELESTRES calculations have shown excellent agreement with fission-gas release measurements [4,5]. In recent years, a number of features have been added to facilitate simulation of high-burnup fuel. These features include: an updated model to reflect more recent data on the degradation of UO_2 thermal conductivity with burnup; variation of UO_2 diffusivity with burnup; variation of fission-gas bubble density with burnup; circumferential cracking of the pellet; larger range for the influences of burnup, enrichment, and element diameter on flux depression; an updated model for solid fission-product swelling; incremental densification; rim effect; and a two-dimensional model for heat transfer between the pellet and the cladding [5].

ELOCA is a computer code developed to model the thermo-mechanical response and associated fission-gas release (FGR) behaviour of CANDU fuel elements during high-temperature LOCA-type transients [6-9]. ELOCA models the stored heat in the fuel, heat generation due to nuclear reactions and heat released from the oxidation of the Zircaloy fuel cladding. Diffusion of heat is followed from the UO_2 fuel to the outside of the fuel cladding, accounting for phenomena associated with changes in thermal properties with irradiation history, effects of fission-gas pressure on fuel-to-cladding heat transfer, pellet eccentricity, and feedback between fuel and cladding deformation and heat transfer. ELOCA is capable of performing multi-segment thermo-mechanical analysis of a CANDU fuel element, accounting for axial variations in thermal hydraulic conditions, element power, Zircaloy microstructure, fuel physical state, fuel-to-cladding heat transfer, and cladding oxidation and deformation. ELOCA currently employs the FREEDOM transient fission-gas release model to couple the thermo-mechanical behaviour with calculations of grain growth, fuel swelling and the formation, diffusion and release of active and stable gaseous fission products [7,8]. The SOURCE fission-product release code will likely replace the FREEDOM fission-gas release module in future versions of ELOCA. ELOCA has been coupled to the CATHENA thermalhydraulics code and incorporated into the FACTAR fuel and fuel channel thermal-mechanical behaviour code.

The SOURCE code calculates the release of fission products from uranium oxide fuel under normal operating and accident conditions [10-12]. The SOURCE code employs a subdivision of the fuel into basis units. Since the fuel volume changes with temperature and porosity, the mass of uranium initially present in the basis unit serves as the constant parameter for describing the basis unit. SOURCE simulates the transfer of fission products between the following inventory partitions: the fuel grain matrix, the fuel grain boundary, the fuel surfaces, the fuel-clad gap, and that released to the coolant. SOURCE models all of the primary phenomena affecting fission-product release from CANDU fuel under accident conditions, namely:

- Fission yield, decay and transmutation,
- Fission-product diffusion,
- Grain growth,
- Grain boundary sweeping,
- Fission-product redistribution due to matrix stripping (fuel volatilization), UO_2/Zircaloy interaction, UO_2 dissolution by molten Zircaloy, fuel melting, grain boundary separation, and grain boundary coalescence / tunnel interlinkage,
- Fission-product vaporization,
- Gap transport,
- Releases due to rewet, and
- Leaching releases.

SOURCE includes models for the effect of fuel stoichiometric deviation (x in $UO_{2\pm x}$) on the fission-product diffusion coefficient. A comprehensive validation of the SOURCE computer code against experimental data is scheduled for completion in 2001 December.

The current focus is on completing the validation of CANDU fuel behaviour and fission-product release codes to the Canadian N286.7-99 nuclear software quality assurance standard. Previous versions of the fuel safety codes were validated in a less systematic fashion using the available experimental data. The current code validation program, however, is aimed at satisfying all of the requirements of the N286.7-99 standard. Future work will focus on additional code development to address issues identified during validation, improve the transfer of information between the codes, and extend their capabilities to advanced CANDU fuel designs currently under development.

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7.1.3 Development of the SCANAIR Code by IRSN, (France)

The objective of the SCANAIR code is to simulate the thermo-mechanical behaviour of a PWR rod (UO_2, MOX), especially at high burnup level, under RIA conditions. It is developed by IPSN, in collaboration with EDF in the frame of the CABRI-REP Na and future CABRI – WATER Loop programmes and has to be validated on global tests (CABRI, some NSRR tests) as well as on separate effect tests (PATRICIA thermo-hydraulic experiments : work underway, SILENE-RIA future tests, …) for correct translation to reactor case.

The main characteristics of the SCANAIR code is to be able to deal with intimately coupled phenomena occurring during rapid power transients such as thermal, mechanical and fission gas aspects. It includes a 1.5D modelling, starting with an initial rod state given by an irradiation code and a power pulse provided as data.

At the present time a satisfying status of validation against the REP Na tests has been obtained.

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7.1.4 Development of Methods for the Analysis of the Fuel Rod Behaviour in the High Burn-up Regime, (Germany)

The project is carried out by GRS under contract of the Ministry for research, technology and economy (BMWi: contract RS 1117).

The objectives of this project is to provide methods describing the fuel rod behaviour under LOCA and RIA conditions. It is the particular goal that these methods will cope with both the high burn-up regime and to the new fuel rod materials. The developmental steps start from existing codes (TESPA for LOCA, SCANAIR for RIA).

The German licensing criteria for LOCA transients are conform with the US criteria except the additional criterion which requires a limitation of the number of bursting fuel rods. The number of burst fuel rods during a LOCA may not exceed 10% of the total number of fuel rods in a core. The code TESPA is regularly used for the determination of the extent of the fuel rod burst for each new core loading. Models which are affected by both the high burn-up (e.g. heat conduction model for UO_2) or by the utilisation of new material (e.g. new cladding alloy) have been identified. The revision and modification of these code models is under way. Furthermore, as far particular high burn-up effects such as transient fission gas release are not explicitly taken into the failure threshold determination, the development and the implementation of related models is initiated.

The present TESPA code recently received a new model which additionally allows the consideration of the behaviour of an internal corrosion layer. Generic LOCA analyses have been performed showing the relative importance of the ductility criteria (PCT = 1200°C and ECR = 17%) with respect to the 10% fuel rod burst criterion. The ductility criteria would allow an operational pin power far beyond that operational pin power which is associated with the 10% criterion. Thus the 10% criterion is still much more restrictive for operational pin power in the high burn-up regime than these ductility criteria.

The work started on the TESPA code development will be continued. The investigation of high burn-up effects on RIA related transients will be initiated as far as the international CABRI project will commence. The present results are still preliminary and therefore unpublished.

7.1.5 Validation of the Fuel Analysis Code FEMAXI-JINS by PIE, (Japan)

Development of original code was performed by JAERI, and the modification of code is by NUPEC/INS sponsored by METI. The objective of the development programme is to verify the accuracy of the code for the evaluation of the integrity of high burn-up fuels, in normal operation and transients of light-water reactors.

The FEMAXI-JINS code is based on the FEMAXI code [1], [2] developed by JAERI. The present status of work is as follows.

- Comparison of the calculated results with the PIE results of high burn-up fuels, for both PWR and BWR, has been performed for inner-gas pressure, fission gas release rate (FGR), diameter change of fuel cladding, etc.
- Some modifications for high burn-up and MOX fuels have also been performed, i.e. rim effect, cladding waterside corrosion (oxidation) model and the power density profile model in the radial direction of fuel pellet.
- Sensitivity study on the primary parameters has been performed to quantify the uncertainty of calculated results by the code.

The progress to date can be summarized as follows:

- For burn-up up to 60 GWd/t, both in PWR and BWR, the code has been verified in respect to fuel thermal conductivity, FGR, fuel swelling, cladding oxide thickness, etc.
- Change of pellet radial power distribution profile with burn-up was taken into account by incorporating the results of burnup calculation code (RODBURN).
- By sensitivity study, the scope of the applicability of model and input parameters of the code has been determined.
- Transient behaviour such as post-BT phenomenon of BWR fuel rod was analyzed by using the results from thermal-hydraulic code.

Future plan include:

- Further study on the validation of FEMAXI-JINS by new PIE results
- Enhancement of model capability for high burn-up fuels
- Statistical evaluation of code uncertainty

References

[1] T. Nakajima and H. Saito, ”A comparison between fission gas release data and FEMAXI-□ calculations”, Nucl. Eng. Des 101(1987)267

[2] M. Suzuki and H. Saito, “Description and User’s Manual of Light Water Reactor Fuel Analysis Code FEMAXI-□(Ver.2)”, JAERI-Data/Code 97-010

7.1.6 The Development of Fuel Safety Analysis Codes in JAERI, (Japan)

This code development exercise in the Fuel Safety Research Lab at JAERI is sponsored by MEXT. Table 7.1.1 gives the state of fuel performance codes in Japan, whilst safety analysis codes are given in Table 7.1.2.

The FEMAXI code will be applied for the safety examination on fuel performance. FEMAXI-V has been completed and registered to NEA Data Bank, whilst FEMAXI-6 is being developed and is scheduled for completion in 2002.

The FURBEL code under development will perform analysis on the effect of axial restraint condition imposed on high burnup fuel cladding, which has different mechanical properties in the radial direction, during reflood of LOCA to evaluate the integrity of fuel rods against axial tensile stress. Completion of the code is expected to take a few years.

In addition, the conceptual design of a new RIA code is in progress. Code development started in 2001and it will be applied for the analysis on the failure mechanism of highly exposed cladding which has different mechanical properties in the radial direction.

References

[1] M.Suzuki, "Analysis of high burnup fuel behaviour in Halden reactor by FEMAXI-V code", Nucl.Eng.Design,vol. 201, pp.99-106, (2000).

[2] S.E.Lemehov, J.Nakamura and M.Suzuki, “PLUTON: A Three-Group Model for the Radial Distribution of Plutonium, Burnup and Power Profile in Highly Irradiated LWR Fuel Rods”, Nuclear Technology, col.3, No.2 pp.153-168 (2001)

[3] M.Suzuki, “Light Water Reactor Fuel Analysis Code FEMAXI-V (Ver.1)”, JAERI-Data/Code 2000-030 (2000).

[4] S.E.Lemehov and M.Suzuki, “PLUTON-Three Group Neutronic Code for Burnup Analysis of Isotope Generation and Depletion in Highly Irradiated LWR Fuel Rods”, JAERI-Data/Code 2001-025 (2001)

Table 7.1.1 Fuel Performance Codes of JAERI for Safety Analysis

Class	Category	Major Phenomena to be Analyzed	Features of Models	Analysis Codes	
				PWR	BWR
Normal Operation and Anticipated Transients	Fuel Performance	Pellet temperature Fission gas release PCMI/ridging deformation Cladding waterside corrosion Cladding creep Pellet swelling, densification Gap size change High burnup fuel behavior	- 2-D non-linear FEM analysis for local PCMI and full length rod deformation, - Burnup dependent thermal conductivity model for pellet, - Model of FP gas release taking into account effect of restraint by stress in the pellet, - Results of neutronics code is incorporated in the calculation. - Verified by Halden test irradiation data, etc. - The code package has been released with a detailed description and I/O manual.	Completed and Released - FEMAXI-IV(Ver.2) - FEMAXI-V --Being Developed -- - FEMAXI-6	
	Neutronics	Pellet power generation density and burnup profiles in the radial direction, Fission products and Pu buildup profile in the radial direction, Power profile in the axial direction, Fast flux profile in the axial direction.	- PLUTON can calculate radial profile of power in UO2 and MOX fuel pellets in any type of water reactor.	Simple code: RODBURN Newly developed: PLUTON	
Accidents	LOCA	Cladding temperature Cladding expansion/ballooning Cladding rupture Cladding axial stress Cladding oxidation	- 2-D non-linear FEM analysis for radial deformation and full length rod deformation, - Oxidation model by oxygen diffusion equation - Reflooding heat transfer	--Being Developed - - FURBEL	
	RIA	Pellet temperature/deformation Cladding temperature Cladding expansion/deformation/rupture Cladding oxidation Mechanical force generation	- 2-D non-linear FEM analysis for radial deformation and full length rod deformation, - Abrupt FP gas release.	-- Planned --	

Table 7.1.2 Safety Analysis Codes of NUPEC/INS

Class	Category	Major Events to be Assessed	Criteria	Analysis Codes: PWR	Analysis Codes: BWR
Normal Operation	Fuel Behaviour			- FEMAXI-JINS	
Normal Operation	Reactor Statics			- CASMO-4/SIMULATE-3	
Normal Operation	Reactor Kinetics	- Global stability - Regional stability	(1) Decay ratio: lower than 1.0		- SKETCH-INS/ TRAC-BF1
Abnormal Transients	Thermal-Hydraulic and Neutronics	- Abnormal control rod withdrawal during start-up and operation - Abnormal boron dilution in reactor coolant - Partial loss of reactor coolant flow - Inadvertent start-up of shutdown loop of reactor coolant system - Loss of off-site power - Loss of main feedwater flow - Loss of load - Others	(1) Mini. Critical Heat Flux Ratio or Mini. Critical Power Ratio: not exceed the limit (2) Fuel cladding : not mechanically fail (3) Fuel enthalpy : not exceed prescribed limit (4) Max. pressure to the pressure boundary : lower than 1.1 times of the design pressure	- RELAP5/MOD2 - COBRA-□-I/ JINS - TOODEE2-JIN S/1	- RELAP5/MOD1 /JINS/B - RELAP4/MOD6 /U4/J3 - SKETCH-INS /TRAC-BF1
Accidents	Thermal-Hydraulic and Neutronics	- Loss of reactor coolant (LOCA) - Loss of reactor coolant flow - Reactor cooling pump failure - Main feedwater pipe rupture - Main steam pipe rupture - Others	(1) The core not seriously damaged and coolable (2) Fuel enthalpy : not exceed prescribed limit (3) Max. pressure to the pressure boundary : lower than 1.2 times of the design pressure (4) Max. pressure to the containment boundary : lower than design pressure (5) No undue radiation exposure to the neighbouring public	- WREM-JINS/P1 RELAP4/MOD5/U2/ J1 TOODEE2-JINS/1 REFLA-1DS REFLA-EM - TRAC-PF1/ MOD2	- WREM-JINS/B RELAP4/MOD6 /U4/J3 MOXY-MOD32 - SKETCH-INS /TRAC-BF1
Accidents	Reactor Kinetics	- Control rod drop (RIA, BWR) - Control rod ejection (RIA, PWR)		- EUREKA-JINS/S	

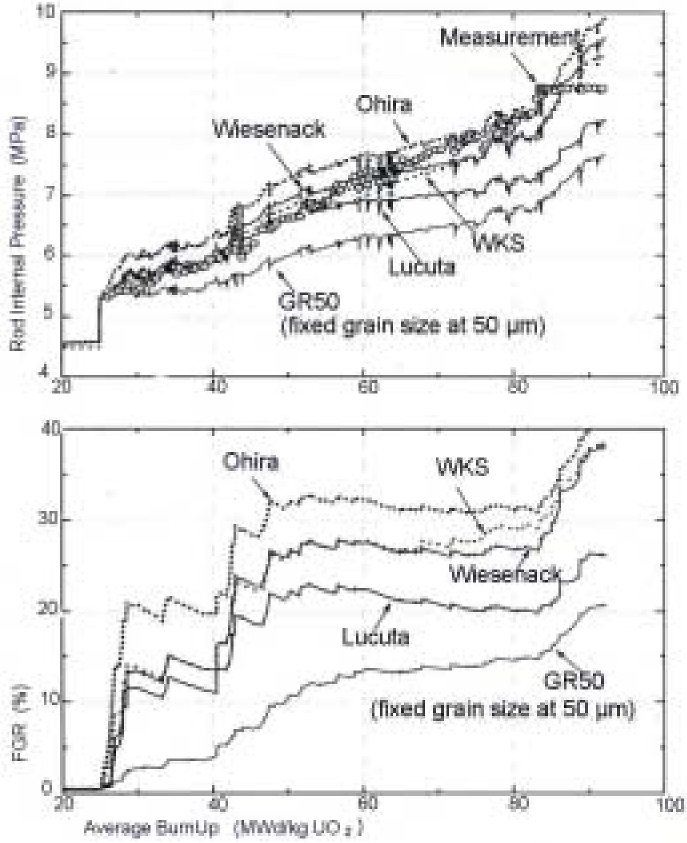

Fig. 7.1.1 Comparison of rod internal pressure (A) and calculated FGR (B) of the rod DH with 100 µm initial diametral gap. Calculations are made with the thermal conductivity models of Wiesenack, Ohira, and Lucuta. WKS denotes the case with Wiesesnack's thermal conductivity and Kosaka's swelling model. GR50 denotes the case with fixed fuel grain size of 50µm and with Wiesenack's thermal conductivity model.

7.1.7 Development of the Code INFRA at KAERI, (Korea)

The Korea Atomic Energy Research Institute is currently developing the INFRA code to predict the transient behaviour of fuel rods. The code embodies a two dimensional finite element method of calculation. Completion of the code is planned for 2003.

7.1.8 Safety Related Fuel Research at the Paul Scherrer Institute, (Switzerland)

The goal of PSI is to contribute to the safety related fuel research with both modelling and experimental work. The goal of the RIA-related research at PSI is to establish a computational base for a new definition of the licensing limits (enthalpy vs. burnup) and to support the licensing body and the utilities with safety-relevant calculations.

Neutronic codes as well as fuel performance codes (FREY [1] and TRANSURANUS, TU [2]) have already been used at PSI for RIA modelling.

For a prototypical PWR RIA case a comparison of results obtained from FREY and TU was performed to assess the merits and shortcomings of the two codes. The base irradiation was modelled with TU and the respective data was transferred to FREY. The fuel pin behaviour during the RIA was then calculated with both TU and FREY. The results were compared and consequences for future RIA-modelling have been derived.

The differences in the calculated fuel centre temperatures between the codes FREY and TU were significant and lie in the range of 50°C. The cladding deformation and loading showed only minor differences. In both calculations, neither fuel melting nor cladding damage was detected for this 1.75$ RIA test calculation.

For the future, a new fission gas model will be implemented in TU by PSI [4]. With this model, it should be possible to model the intergranular swelling during the RIA as well as a possible burst release. The core simulators will generate the necessary power histories at several burnup steps.

References

[1] "FREY-01: Fuel Rod Evaluation System", EPRI NP-3277, volumes 1 and 2, Revision 3, Final report, Electric Power Research Institute (EPRI), August 1994.

[2] K. Lassmann, „TRANSURANUS: a fuel rod analysis code ready for use“, J. Nucl. Mater. 188 (1992) 295.

[3] F. Holzgrewe, D. Gavillet, R. Restani, M.A. Zimmermann, H.U. Zwicky, P. Jourdain: “Validation of CASMO-4 against SIMS measured spatial nuclide distributions inside a 9 wt% Gd BWR pin”, Proceedings of the 2000 International Topical Meeting on LWR Fuel Performance, Park City, Utah (US), April 10th –13th, 2000.

[4] H. Wallin, L.Å. Nordström, Ch. Hellwig, "Fission Gas Model of the Fuel Code SPHERE-3", Proc. IAEA/NEA/CEA International Seminar on Fission Gas behaviour in Water Reactor Fuels, Cadarache (F), Sept. 2000.

7.1.9 Modelling Hydride Embrittlement and Fracture in Hydride-Forming Metals, (Sweden)

The objectives of this work are to model the mechanisms of hydrogen-assisted cracking in Zr-based alloys. The work is performed by the Malmö University and sponsored by the Swedish state and the Swedish Nuclear Industry Research Programme/Fuel Group (BFUK/PGB)

Theoretical model and finite element algorithm for steady-state crack growth due to delayed hydride cracking has been completed. Finite element modelling of mechanical testing methods for cladding is also included. Additional modelling refinement and extension is ongoing. A theoretical model and finite element algorithm for steady-state crack growth due to delayed hydride cracking in material with a hydrogen concentration gradient has been completed. Finite element modelling of mechanical testing methods for cladding is also included. Further refinement and extension of the modelling is planned.

References

[1] A.G. Varias, P. Ståhle, A.R. Massih and R. Warren
KKS Project "Hydride embrittlement and fracture in hydride-forming metals".
Annual Progress Report 2001
Dnr 277/00, Malmö University

8. Thermal Hydraulics Studies

8.1.1 Thermal Hydraulic Test for PWR Fuel Assemblies, NUPEC-TH-P Project, (Japan)

This work is carried out in the DNB Test Facility of PWR Fuel Assembly at the Takasago Engineering Laboratory of NUPEC (Hyogo/Japan), Figure 8.1.1, sponsored by METI. The objective of the studies is to prove the reliability and the integrity of the PWR fuel assemblies by carrying out the DNB test, in-bundle void fraction test and post DNB test.

The following thermal hydraulic tests have been conducted from 1987 to 1994:

- Critical heat flux test
- In-bundle void fraction test

Post DNB tests started in 1995 and have been completed in 2002

Evaluation of the experimental work includes:

- Evaluation of the current PWR thermal hydraulic design reliability
- Evaluation of the void correlations and subchannel analysis method on PWR
- fuel assembly
- Evaluation of the post DNB thermal characteristics for PWR Fuel Assemblies

From tests conducted to date, the effect of a bowed rod on DNB heat flux was verified and the current design procedure of rod bow DNB penalty estimation was confirmed to be conservative. (Figure-8.1.2) Reliable void correlations were developed by using gamma-ray transmission methods. (Figure-8.1.3). The effects of various parameters such as grid span length and mixing vane effect on post DNB heat transfer characteristics were surveyed in Post-DNB model test using Freon as working fluid. (Figure-8.1.4).

The mock-up post DNB test using water as the working fluid started in 1999 and is to be completed in March, 2002.

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[1] M. Akiyama et al. : Reliability proven test on maximum thermal loading of PWR fuel assembly, Journal of the Atomic Energy Society of Japan Vol.36,No.1 (1990) (in Japanese)

[2] H. Nishioka et al.: Total evaluation of bundle void fraction measurement test of PWR fuel assembly, ICONE-4 (1996)

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8.1.2 Thermal Hydraulic Test for BWR Fuel Assemblies NUPEC-TH-B Project, (Japan)

This work is carried out in the Thermal Hydraulic Test Facility of Isogo Engineering Laboratory of NUPEC (Yokohama/Japan), sponsored by METI, Figure 8.1.5.

The objective of the test is to prove reliability and integrity of the BWR Fuel Assemblies, by carrying out the transient critical power test and the flow induced vibration test etc. The assemblies are 8x8 fuel and 9x9 fuel, (9x9 fuel has type A and type B fuel that are different with design).

Main specification	8x8	9x9 type A	9x9 type B
1.Fuel bundle Number of rods Max. bundle burn-up (GWd/t)	60 50	74 (include partial length rod 8) 55	72 55
2.Water rod Number of rod Shape of cross section	1 Circle	2 Circle	1 Square

For Thermal Hydraulic Testing, the following tests have been conducted from 1991 to 1999:

- Steam void fraction measurement test
- Critical power test
- Post boiling transition (BT) test
- Flow induced vibration test
- Pressure drop measurement test

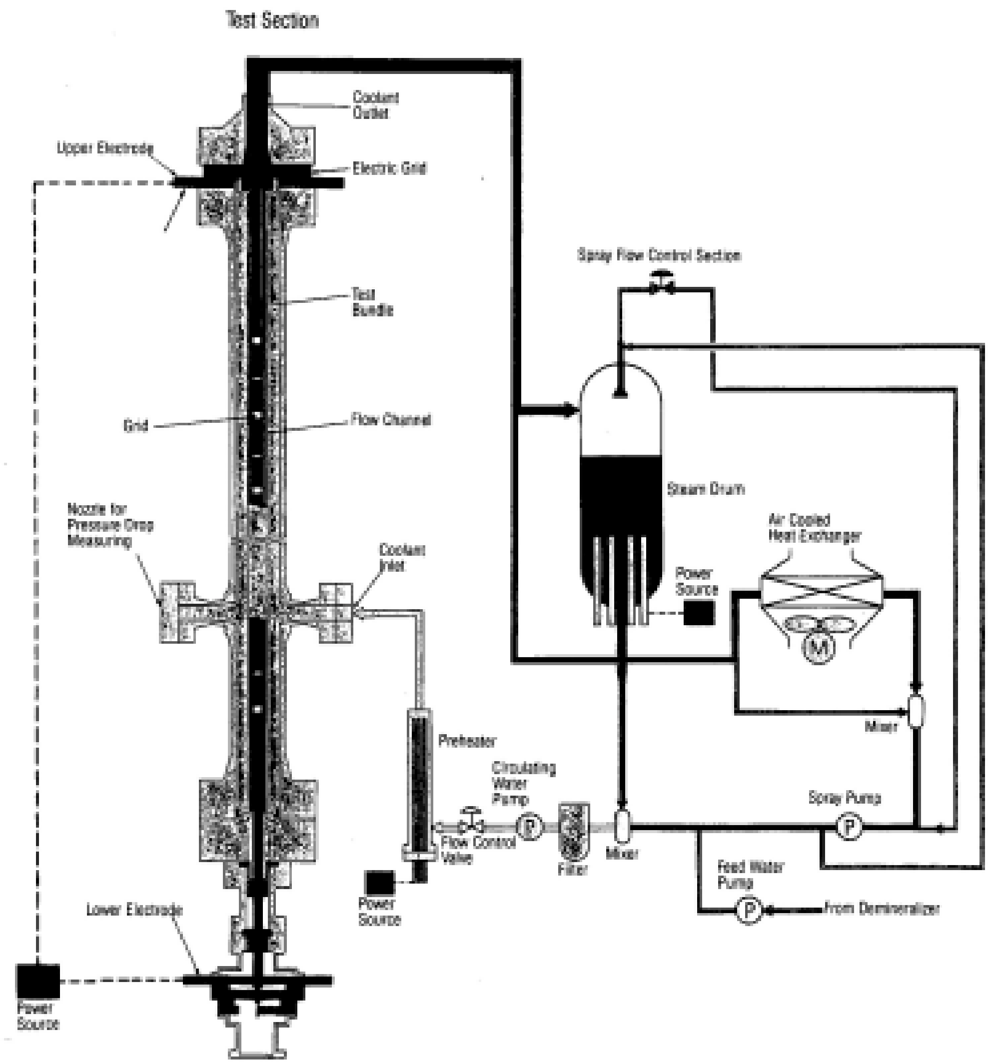

Fig. 8.1.1 System Diagram of The Test Facility

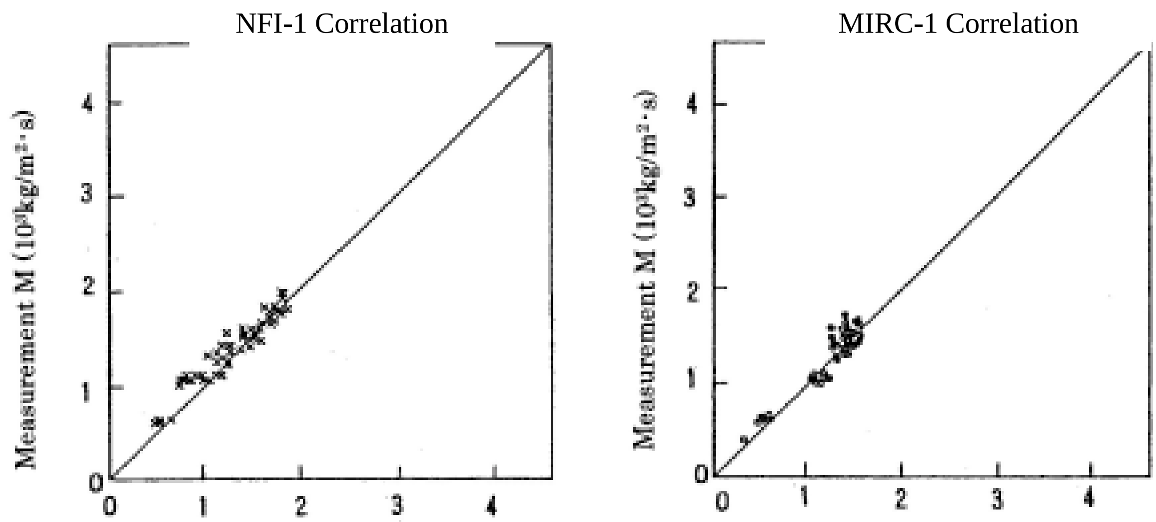

Fig. 8.1.2 The Applicability of Critical Heat Flux Correlations

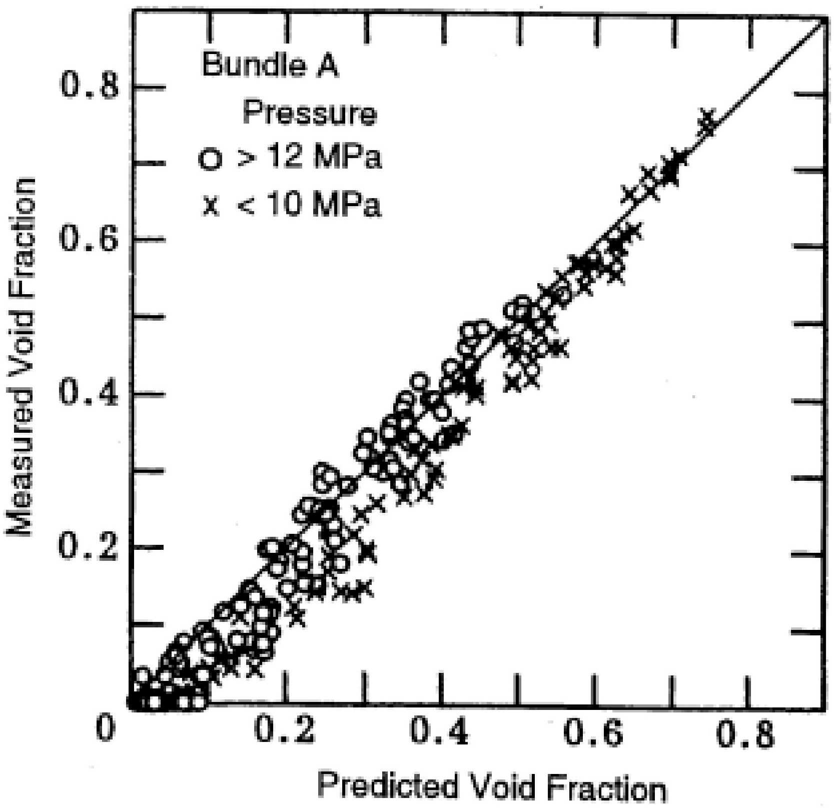

Fig. 8.1.3 Comparison of Rod Bundle Data with Predictions

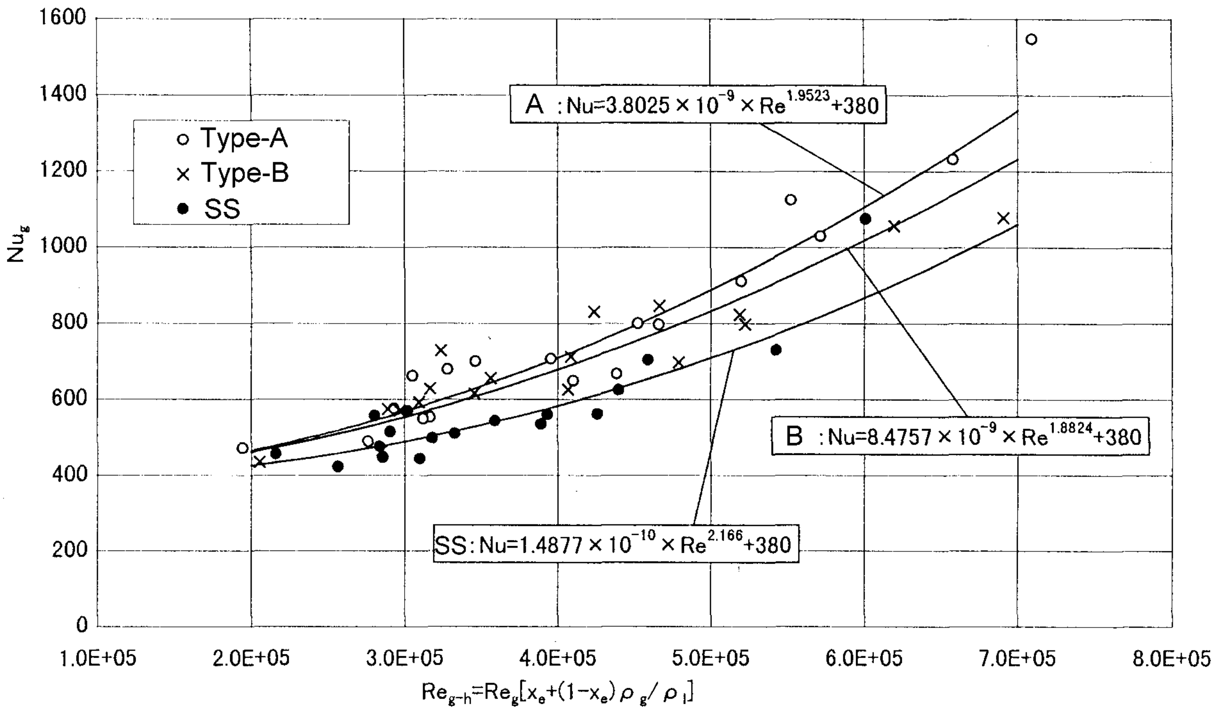

Fig. 8.1.4 Effect of grid spacer type, Type-A/Type-B and SS, on film boiling heat transfer

Evaluation has been performed to evaluate the current BWR thermal hydraulic design reliability:

- to develop new correlations that predict void fraction.
- to develop new correlations that predict post BT heat transfer coefficient and rewet quality.
- to develop new correlations that predict fuel rod vibration amplitude.
- The current BWR design correlations have been proven to agree well with the measurements. Figure8.1.6 shows the comparison of critical powers calculated by the design correlations and the measured for 9x9 fuel.
- Reliable void fraction data were obtained by using an X-ray computed tomography scanner. Figure 8.1.7 shows typical local void fraction distribution for 8x8 fuel.
- The cladding temperature rises after BT, however, the rising rate becomes small due to power decrease.
- The temperature returns to initial level due to rewet after a few seconds. Figure 8.1.8 shows cladding temperature behavior during flow decrease transient for 9x9 type B fuel.
- The vibration characteristics were evaluated and confirmed to be good agreement with experimental results. Figure 8.1.9 shows the comparison of rod displacements calculated by the FEM analysis and the measured for 9x9 type A fuel.

The project has been completed in March 2001.

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[1] A.Inoue et al.: Void fraction distribution in a boiling water reactor fuel assembly and the evaluation of subchannel analysis codes, NUCLEAR TECHNOLOGY Vol.2 (1995)

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[4] Nuclear Power Engineering Corporation: Annual Report 2000 (2000)

[5] M.Akiba et al.: Thermal hydraulic test for BWR 9x9 type fuel assemblies (3)-(5), The Atomic Energy Society of Japan (1998) (in Japanese)

[6] M.Akiba et al.: Thermal hydraulic test for BWR 9x9 type fuel assemblies (8) -(9), The Atomic Energy Society of Japan (1999) (in Japanese)

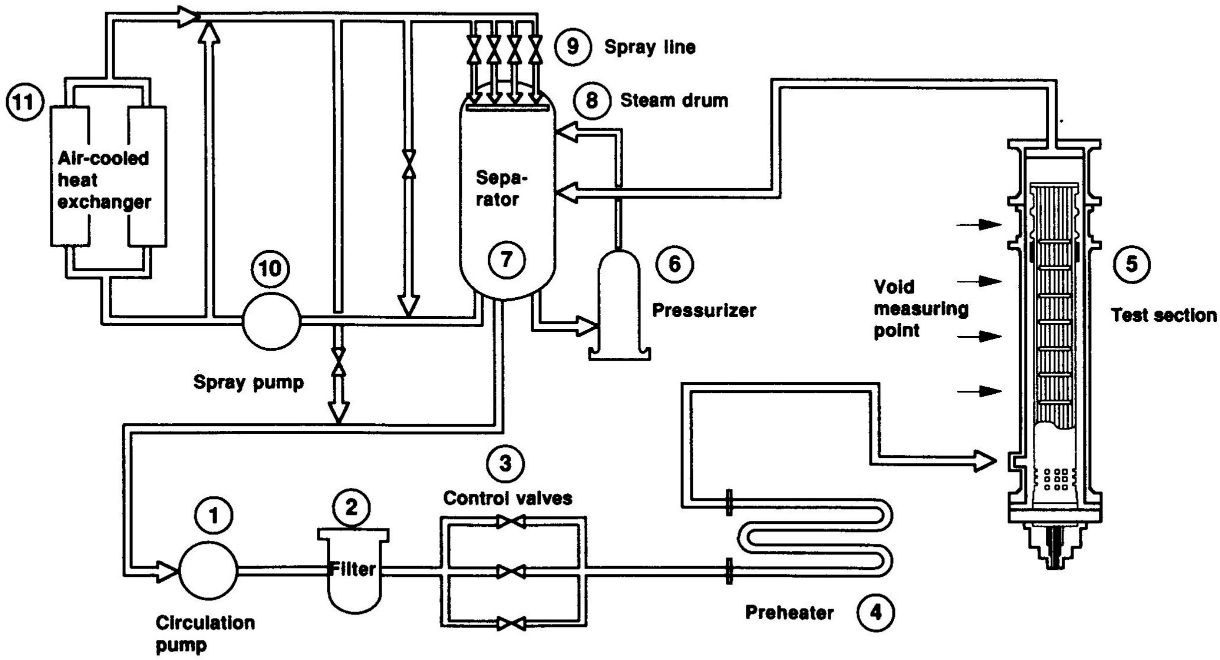

Fig. 8.1.5. System diagram of the test facility

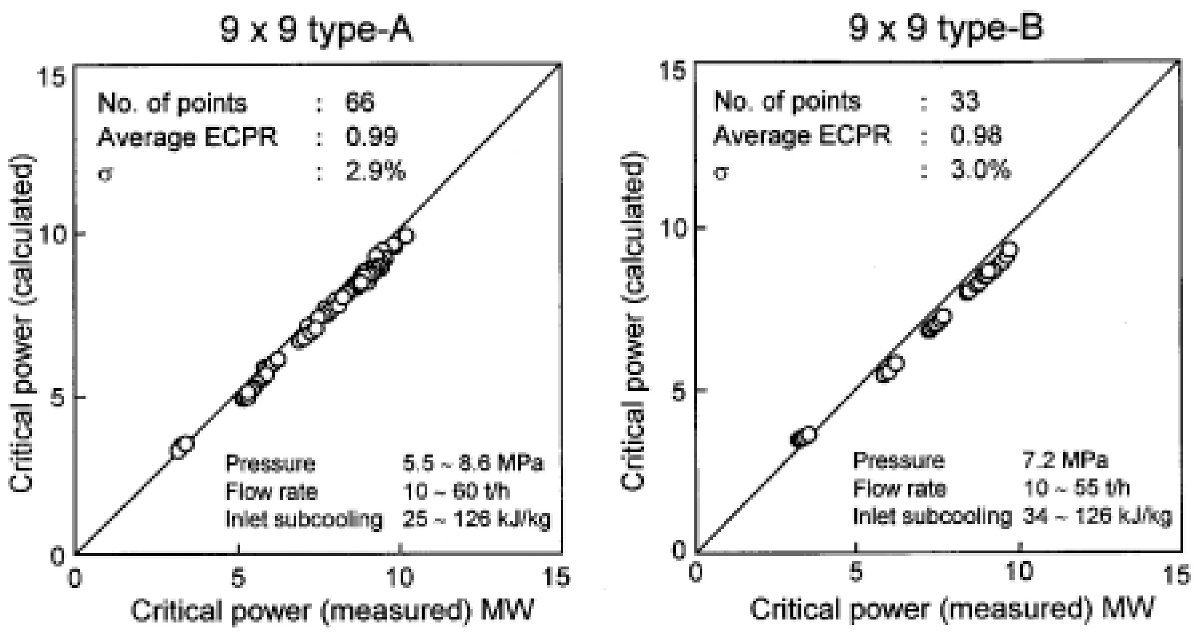

Fig. 8.1.6. Comparison of critical power with calculation for 9x9 fuel

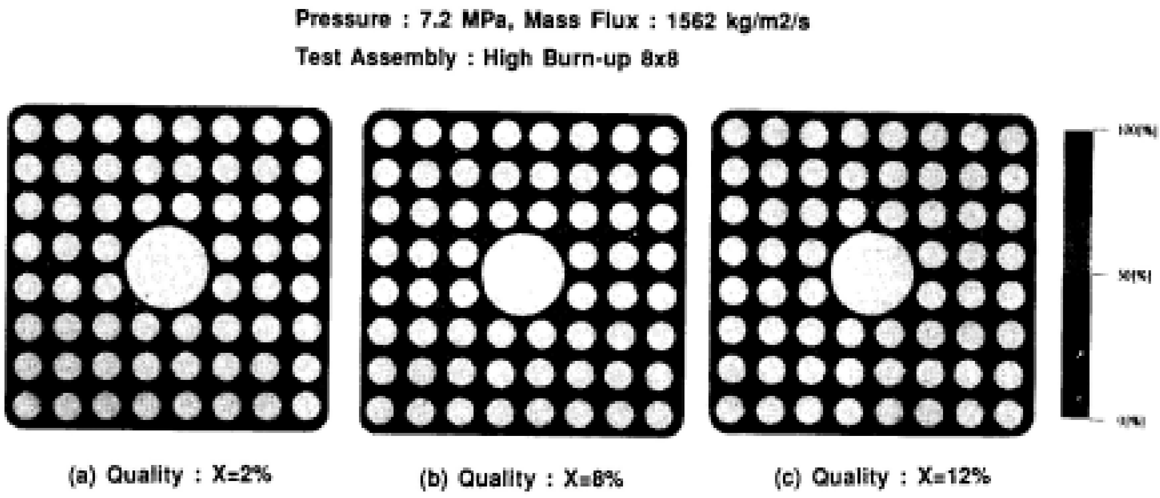

Fig. 8.1.7. Typical local void fraction distribution by X-ray CT scanner for 8x8 fuel

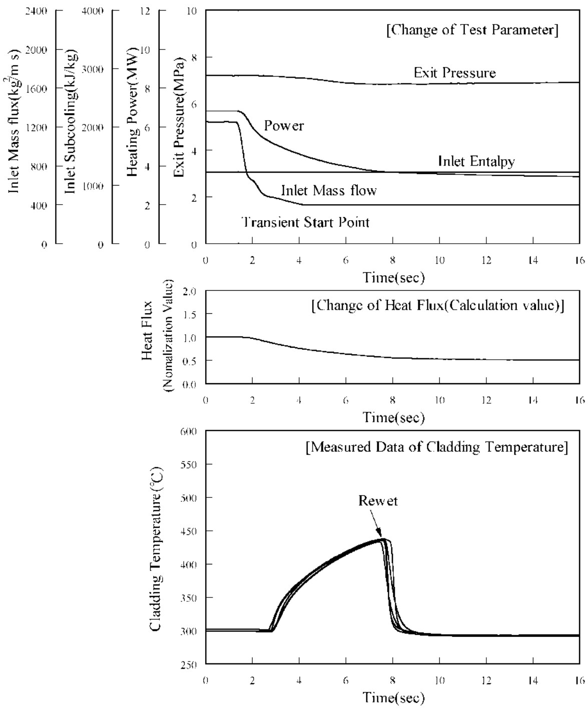

Fig. 8.1.8. Cladding temperature behavior during transient for 9x9 type B

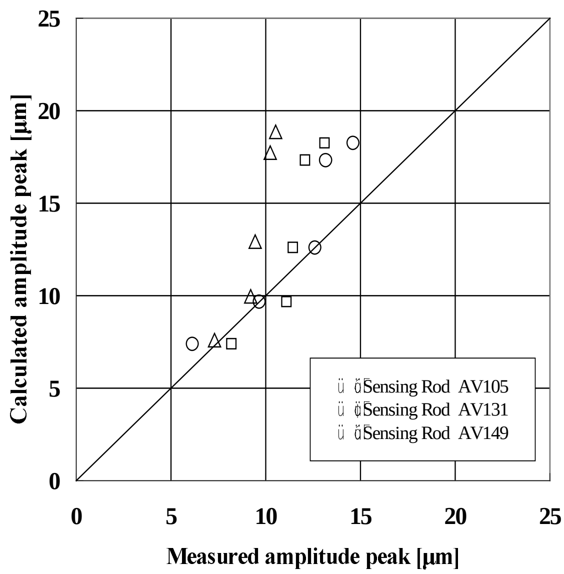

Fig. 8.1.9. Comparison of the amplitude peak with calculation for 9x9 type A fuel

9. Reactor Physics Codes

9.1.1 Reactor Core Analysis Code System CASMO-4/SIMULATE-3, (Japan)

The original code was developed by Studsvik of America, Inc and was purchased by NUPEC/INS (Nuclear Power Engineering Corporation/Institute of Nuclear Safety). The development is sponsored by METI. The purpose is to calculate and evaluate the nuclear characteristics of PWR and BWR, such as power distribution, burn-up-distribution, reactivity worth, core-wide reactivity coefficients and kinetic parameters.

CASMO-4 is a multi-group two-dimensional transport theory code for burn-up calculation on BWR and PWR assemblies or simple pin cells. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with gadolinium, burnable absorber rods, cluster control rods, in-core instrument channels, water gaps and cruciform control rods in the regions separating fuel assemblies. Nuclear data are collected in the library containing microscopic cross sections in 70 energy groups. Two-group cross sections of the assembly and discontinuity factors are calculated for use in reactor core simulator : SIMULATE-3.

SIMULATE-3 is an advanced three-dimensional two-group nodal code for the analysis of both PWRs and BWRs. The code employs analytic functions (sinh and cosh) to model steep intra-nodal flux distributions in both the fast and thermal groups. The types of calculations performed by the code are:

- Depletion in three dimensions, 1/8, 1/4, 1/2 or full core.
- Reload shuffling including reinsertion of discharged fuel
- Reactivity coefficient calculations
- Rod worth, including shutdown margin
- Xenon transients
- Core follow
- Start-up predictions
- Criticality searches

The code verification against cores containing MOX fuel was performed using the following critical experiment data and reactor core data:

- VIP-BWR critical experiment containing MOX fuel [1].
- EPICURE critical experiment containing MOX fuel [2].
- Tihange unit-2 reactor core data (3-loop PWR containing MOX fuel).
- MISTRAL critical experiment containing MOX fuel in the full core [3].

The code capabilities of burnup calculations will be checked against the PIE data of UO_2 and MOX fuels irradiated in BWR and PWR.

References

[1] "Verification of Reactor Analysis Code for Core containing MOX Fuel, Analysis of VIP-BWR Critical Experiment ", INS/M96-06, NUPEC, March, 1997.

[2] "Maintenance of Reactor Analysis Code , Analysis of MISTRAL and EPICURE Critical Experiments", INS/M98-29, NUPEC, March, 1999.

[3] "Maintenance of Reactor Analysis Code, Analysis of MISTRAL Critical Experiment (Full MOX Core)", INS/M99-27, NUPEC, March, 2000.

9.1.2 3D Netronic Transient Code: EUREKA-JINS/S, (Japan)

Development of the original code was by JAERI with modifications made by NUPEC/INS under the sponsorship of METI. EUREKA-JINS/S is a three dimensional neutronic code to analyze RIA, such as control rod drop accidents in BWR and control rod ejection accidents in PWR. Its purpose is to calculate the fuel enthalpy distribution in the core and the number of fuel failures during transients.

Three dimensional neutronic transient code "EUREKA-JINS/S" [1] is a modification of EUREKA-SPACE [2] code originally developed by JAERI. In the EUREKA-JINS/S code, the model to calculate the reactivity insertion due to control rod movement and the feedback due to changes of fuel temperature and moderator density is completely revised by applying the perturbation theory.

The code uses a quasi-static kinetics model coupled with a multi-channel thermal-hydraulic model based on a homogeneous flow model. In the quasi-static kinetics model, the space-time neutron kinetics is described by assuming variable separation between the amplitude function and the shape function. The amplitude function is described by point kinetics equations for up to six delayed neutron precursor groups indicating the overall dynamic behaviour of core power. The shape function is described by two-group diffusion equations indicating the spatial power distribution in the core. In calculating the shape function, three-dimensional diffusion equations are solved by using CITATION code [3].

The thermal-hydraulic model in EUREKA-JINS/S is based on 3-balance equations for a homogeneous flow. In the thermal-hydraulic calculation, the core is modeled with multiple parallel coolant channels and each channel is divided into a number of axial nodes. The thermal heat transfer from the fuel to the coolant is calculated by solving the radial one-dimensional heat conduction equation with the Crank-Nicholson method for each axial node of each channel. The pressure drop across the channel is calculated by solving the momentum equation with the Newton-Raphson iterative procedure.

The major development of code is completed. Now, EUREKA-JINS/S is used to review the licensing analysis of RIA at NUPEC/INS to support the safety examination conducted by METI. The recent review analysis of RIA is as follows:

- Control rod ejection accident in PWR (Takahama-3,4) containing MOX fuel (1998).
- Control rod drop accident in BWR (Fukusima-1) containing MOX fuel (1999).
- Control rod drop accident in ABWR (Ouma) containing MOX fuel in the full core (2000).

At present, the verification of EUREKA-JINS/S is performed against the reactivity accident test results conducted with the SPERT reactor in USA.

References

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9.1.3 Development of the 3D Neutron Kinetics and Thermal Hydraulics Analysis Code: SKETCH-INS/TRAC-BF1, (Japan)

Development of the original code was made by JAERI and INEL (Idaho National Engineering Laboratory). Modifications were made by NUPEC/INS sponsored by METI.

Three-dimensional neutron kinetics and thermal-hydraulics analysis code "SKETCH-INS/TRAC-BF1" [3] is a modification of TRAC-BF1/SKETCH-N [(4] code originally developed by JAERI coupling the transient analysis code TRAC-BF1 with the three-dimensional neutron kinetics code SKETCH-N. The coupling interface module is based on the massage-passing library of Parallel Virtual Machine (PVM).

SKETCH-INS solves neutron diffusion equations in 3D XYZ coordinates for steady-state and kinetics problems. The code treats two neutron energy groups and six delayed neutron precursors. The nodal method is used for spatial integration of diffusion equations. Time integration of the neutron kinetics equation is performed by the fully-implicit scheme with an analytical treatment. The homogenization method using the assembly discontinuity factor is implemented in the code to improve the calculation accuracy for heterogeneous core.

TRAC-BF1 [5] is a version of the TRAC code modified in INEL for BWR analysis. A hydrodynamics model of the code is a two-phase two-fluid model. Six conservation equations for mass, momentum and energy are formulated for vapor and liquid phases for one-dimensional flow. Fuel rod heat conduction module solves 1D radial beat conduction equations. Heat transfer at the outer cladding surface is modeled separately to liquid and gas phases. Heat transfer curve is subdivided into single-phase liquid, nucleate boiling, transition boiling, film boiling, single-phase vapor and condensation modes. Semi-implicit method is applied in time integration of the fluid flow equations. Finite-different methods are used for space integration of the both fluid and heat conduction equations.

The current application of the code is BWR stability analyses to evaluate the stability parameters of decay ratio and resonance frequency for both the core-wide oscillation and the regional oscillation.

The major modification of code has finished. Now, SKETCH-INS/TRAC-BF1 is used for the licensing review on the BWR stability analysis at NUPEC/INS to support the safety examination conducted by MITI. A recent review analysis was performed on the stability of ABWR (Ouma) with MOX fuel in the full core (2000).

The verification of the code [1] has been carried out for the OECD/NEA benchmark [2] concerning to the BWR stability data by the Ringhals 1 reactor in Swedish. Further verification of the code is planned for the OECD/NRC benchmark concerning to the BWR turbine trip data by the Peach Bottom 2 reactor in USA.

In the future, extension of the code's applicable range to BWR transient analyses ranging from plant abnormal transients to accidents, such as RIA.

References

[1] "Modification of Three Dimensional Plant Transient Analysis Code TRAC-BF1/SKETCH-N: Ringhals 1 Stability Analysis", INS/M99-13, NUPEC, March, 2000.

[2] Lefvert, T., OECD/NEA Nuclear Science Committee BWR Stability Benchmark, Final Specifications", NEA/NSC/DOC(94)15, 1994.

[3] "Modification of Three Dimensional Plant Transient Analysis Code TRAC-BF1/SKETCH-N", INS/M99-12, NUPEC, March, 2000.

[4] Asaka H., V.G. Zimin, Iguchi T. and Anoda Y.: Coupling of the Thermal-Hydraulics TRAC Codes with 3D Neutron Kinetics Code SKETCH-N, Preliminary Proc. of the OECD/CSNI Workshop on Advanced Thermal-Hydraulic and Neutronics Codes: Current and Future Applications, Barcelona, Spain, 10-13 April, 2000, vol. 2, pp. 1-15, 2000.

[5] Borkowski, J. A., et al.: TRAC-BF1/MOD1: An Advanced Best-Estimate Computer Program for BWR Accident Analysis, Model Description, NUREG/CR-4356, EGG-2626, Vol. 1, August, 1992.

10. Severe Accident Studies

10.1.1 Fuel Moderator Interaction (MFMI) Experiments, (Canada)

These MFMI experiments will be performed at the Chalk River Laboratories of AECL. The CANDU Owners Group (COG) is sponsoring these experiments under joint funding from Ontario Power Generation, Hydro Québec, New Brunswick Power, and AECL.

The MFMI experiments are being performed to improve our understanding of the energetics of the interaction of molten material with the heavy-water moderator under CANDU single-channel accident conditions, and to provide data for use in reactor safety code validation. The stagnation feeder break and flow blockage scenarios both lead to fuel channel failure and have the potential for injection of small quantities of molten UO_2-Zircaloy mixtures into the heavy-water-filled moderator vessel. The results of the MFMI experiments will be used to verify the assumption currently made in CANDU safety analysis that classical steam explosions do not occur under these conditions.

The main feature of the MFMI facility is a robust confinement vessel (5.5 m tall, 1.5 m in diameter) located inside a concrete pit enclosed in a concrete building. The experimental plan provides for incremental increases in the molten material loading to help manage this risk. The MFMI experimental program has been defined and the facility is under construction; No experimental results have been obtained to date.

11. Conclusions

Normal Operations; UO_2 and MOX fuel

Over the last twenty years there has been a gradual increase in discharge burn-up from commercial power reactors. With the original 3 cycle operation, the discharge burn-up was of the order 30 MWd/kg but now, most countries have increased that to a level approaching 60 MWd/kg, see Table 11.1. This has been accompanied by intense R&D both in test reactors like that operated by the OECD Halden Reactor Project and R2 at Studsvik and LTA irradiation in power reactors.

Table 11.1 Current Burn-up Limits in Various OECD Countries

Country	*Fuel type*	*Limit (MWd/kg)*
Canada	CANDU	20
Czech Republic	VVER, PWR(1)	various spec.
Finland	VVER, BWR	40
France	PWR	52 (UO_2), 42 (MOX)
Germany	PWR/BWR	52 – 57
Hungary	VVER	60 (BNFL), 55 (Russian) (2)
Japan	BWR	55 (UO_2), 40 (MOX)
	PWR	48 (UO_2), 45 (MOX)
Korea	PWR	60 (W), 58 (CE) (2)
	Candu	
Spain	PWR/BWR	various spec.
Sweden	PWR/BWR	various spec.
Switzerland	PWR/BWR	various spec. up to 60 (PWR) and 50 (BWR)
USA	PWR/BWR	various spec. up to 62(2)
Netherlands	PWR	various spec. up to 60

(1) VVER-1000 reactor with Zr-4 fuel cladding *(2) Rod average. All other values are assembly average*

Thus the main goal for the industry is to consolidate the safety issues to bring all countries up to ~60 MWd/kg and possibly 65 MWd/kg. It is to be noted that this is close to the limit achievable with the current restriction of 5% ^{235}U enrichment. To do this, the fuel assessment codes must be well validated with robust models describing all aspects of fuel behaviour. In this respect, regarding UO_2 fuel behaviour, the main aspects are:

- thermal conductivity and its degradation with burn-up
- rod internal pressure; the avoidance of clad lift-off due to high fission gas release, (FGR)
- the effect of the High Burn-up Structure, (HBS), or 'rim' structure on fuel performance; its effect on thermal conductivity and FGR
- fuel swelling and its effect on Pellet Clad Mechanical Interaction, (PCMI).

Regarding cladding, the three challenges to its integrity at high burn-up are:

- waterside corrosion
- degradation of mechanical properties due to embrittlement by irradiation damage and hydrogen pick-up during corrosion
- irradiation growth leading to mechanical interaction with assembly components.

Experiments on these topics are currently underway to augment data already obtained. Also there are developments in materials leading to improved characteristics of both fuel pellet and cladding namely:

- large grain sized fuel to reduce FGR
- 'Soft Pellets' containing additives to produce a more compliant pellet, thus reducing PCMI
- advanced Zr alloys to reduce waterside corrosion and hydrogen pick-up.

It is anticipated that with the current research programmes there are sufficient data to develop and validate fuel performance codes to support these higher levels of discharge burn-up.

The main goal for MOX fuel is for it's safety issues to be treated indistinguishably from those of UO_2. Research to date has shown that MOX pellets have a slightly worse thermal conductivity but similar degradation with burn-up. At the same time, it exhibits slightly greater fission gas release, although the onset of release as a function of temperature and burn-up is little different from that of UO_2. FGR from MOX at high burn-up is exacerbated by a higher reactivity than UO_2 due to its neutronics characteristics. Improvements in MOX are being pursued by vendors by investigating the effect of homogeneity on thermal performance and FGR. However, MOX pellets are naturally more compliant than UO_2 because of their higher rate of thermal creep.

LOCA

The 1980s saw great attention paid to the LOCA scenario and much data are relevant today. However, the data on high burn-up behaviour are rather scarce. In this respect, aspects such as possible fuel relocation or 'slumping' into the ballooned area leading to higher clad temperatures need addressing as well as the effect of axial constraints during quenching. Most important is the need to re-visit the 17% Equivalent Clad Reacted (ECR) criterion in the light of new alloys and new geometries, (clad diameters and thickness, see Table 11.2).

Table 11.2 Examples of PWR and BWR fuel designs

Assembly type	PWR 14 x 14	PWR 17 x 17	BWR 8 x 8	BWR 10 x 10
Cladding outer diameter, mm	10.72	9.50	12.52	9.62
Cladding inner diameter, mm	9.48	8.36	10.79	8.36
Cladding wall thickness, mm	0.62	0.57	0.87	0.63
Cladding cross section, mm^2	19.70	16.00	31.67	17.80
Fuel pellet diameter, mm	9.29	8.19	10.57	8.19
Fuel pellet cross section, mm^2	67.80	52.68	87.75	52.68
Rod pitch, mm	14.10	12.60	16.30	12.40
Rod-to-Rod distance, mm^2	3.38	3,10	3.78	2.78
Water cross section (subchannel) mm^2	108.5	87.9	142.6	81.1
Total clad cross section in assembly mm^2	3861	4624	2027	1780
Cladding surface to fuel volume ratio mm^{-1}	0.50	0.57	0.45	0.57
Cladding-to-fuel cross section ratio	0.29	0.30	0.36	0.34
Water to fuel cross section ratio, subchannel	1.60	1.67	1.63	1.54

Notes:
- Total clad cross section increased in PWR from 14x14 to 17x17, decreased 10% in BWR from 8x8 to 10x10.
- Cladding surface / fuel volume ratio increased in PWR and BWR.
- Cladding to fuel cross section ratio relatively constant.
- Coolant-to-fuel ratio changed by only ~5% in PWR and BWR.

Two aspects of fuel slumping are the impact of fuel-clad bonding on the propensity of slumping and FGR in the slumped region leading to increased local pressure. Another aspect is the impact of axial gas flow through a ‘tight’ fuel column on ballooning behaviour. From section 5.2 it is clear that there are several research programmes addressing the properties of high exposure cladding to LOCA but not so much on separate effects fuel studies. Halden have carried out axial gas flow studies in fuel rods over a range of burn-up and test have shown a severe restriction in volume flow at high burn-up thus restraining the rate of clad ballooning. What is lacking therefore is evidence for or against slumping and the internal pressure generated by FGR within the slumped region during the temperature/time envelope of the transient.

RIA

Early experiments on low burn-up fuel showed that fuel failure by rapid reactivity insertion only occurred after energy depositions around 200 cal/g. It was not until the first CABRI REP Na test on high burn-up fuel with severely oxidized cladding that it was realized that under these conditions a much lower energy deposition caused fuel failure and extensive fuel dispersal. This result initiated a renewed interest in this type of accident with integral and separate effects tests initiated as outlined in section 6. The main issues with respect to the cladding are its mechanical response during high rates of strain and the effect of hydides on the mechanical properties. Regarding the fuel, tests have shown that the clad strain was greater than that expected from thermal expansion of the fuel pellet. Consequently, there would appear to be a new loading force imposed by high burn-up pellets, so the goal is to explain this new force and quantify it. In this respect, the high burn-up structure at the pellet rim is under intense separate effects study as this is anticipated to be the root cause of the increased clad loading. The reduction in acceptible energy deposition at high burn-up is considered therefore to be a result of degraded clad mechanical properties and increased strain from restructured regions of pellets. It is clear that the several research programmes both separate effects studies on fuel, section 6.2 and on cladding, section 6.3 should in the near future lead to a better understanding of this type of accident. As a separate but parallel study, it is important to improve reactor physics codes and calculation to see whether or not it is possible to deposit energies as high as those required for fuel failure.

Resolution of Safety Issues

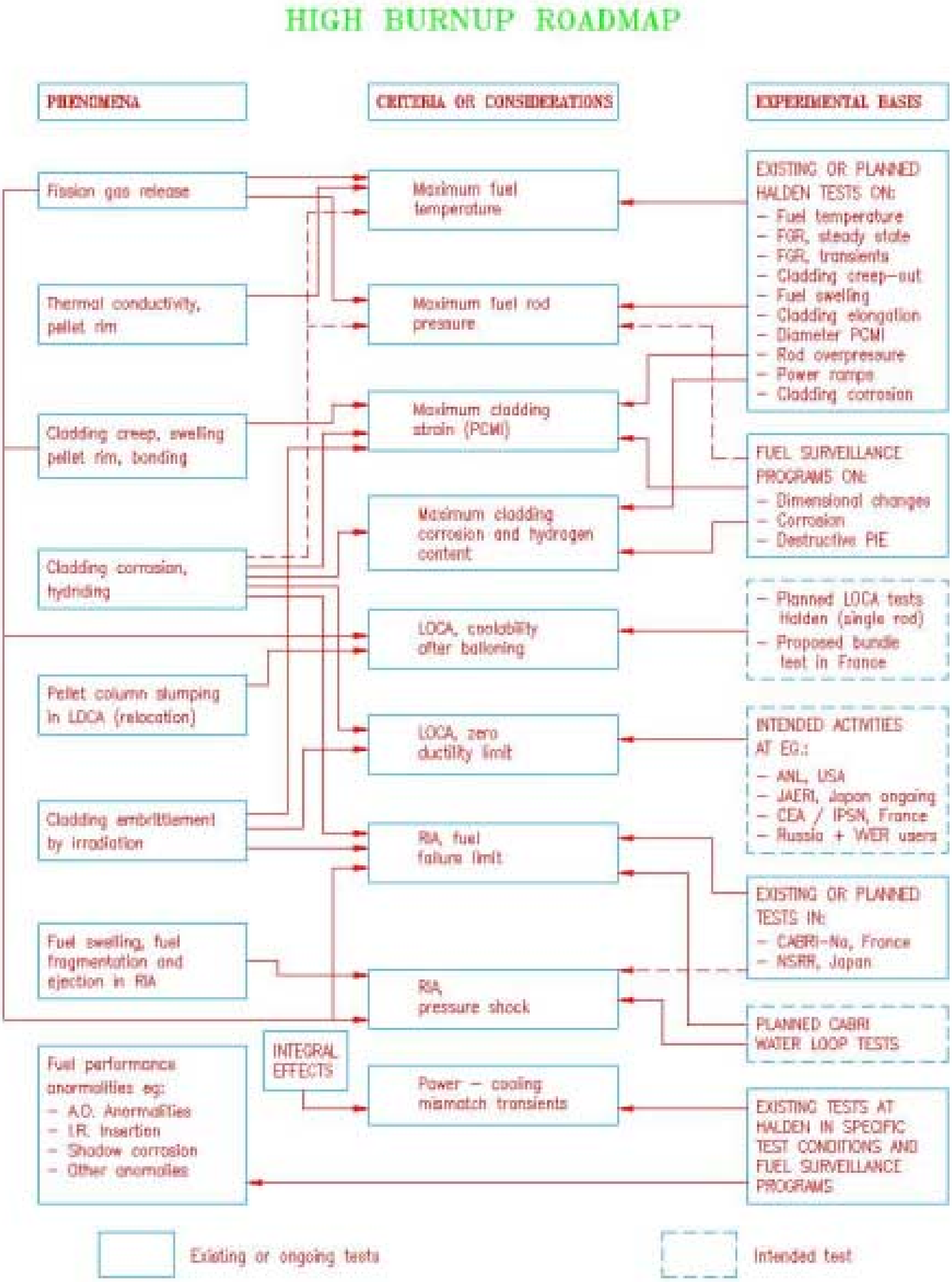
As a final comment on the inter-play between phenomena influencing fuel safety, the criteria used to assess compliance and the experimental database on which such criteria can be derived, Figure 11.1 shows a ‘road map’ linking these three components. From this it is easy to identify where supporting data already exist and where new data will be generated by ongoing programmes or programmes already in the planning stage.

This report has addressed safety issues as they are applied to current reactor systems, the so called ‘Generation II’ designs. There are now advanced reactor designs categorised as generations III, III+ and IV. A common element of these is the introduction of passive safety features. Thus one question which requires consideration is whether or not these new designs will operate within the safety envelope of the current design. If this is the case, then the scenarios currently being addressed should apply without extension to these new systems. Hence, future R&D will concentrate on compliance of new materials to the current or a reduced safety envelope and not the consideration of new scenarios.

It is clear that the currently most important issues to the international nuclear industry are high burn-up performance both in normal operations, LOCA and RIA conditions. The survey of international research programmes outlined above demonstrates the large element of activity to address these issues. When put together, the individual programmes add up to a tremendous effort in both time and money and will ultimately lead to a much better understanding of materials and component behaviour in a wide range of postulated scenarios.

It is very important therefore, that these activities are well supported and that their results should be made available to the widest possible audience. Thus ensuring a common culture of safe and economic production of electricity from nuclear power generation.

Figure 11.1


HIGH BURNUP ROADMAP
PHENOMENA
CRITERIA OR CONSIDERATIONS
EXPERIMENTAL BASIS
Fission gas release
Thermal conductivity, pellet rim
Cladding creep, swelling pellet rim, bonding
Cladding corrosion, hydriding
Pellet column slumping in LOCA (relocation)
Cladding embrittlement by irradiation
Fuel swelling, fuel fragmentation and ejection in RIA
Fuel performance anormalities eg:
- A.O. Anormalities
- I.R. Insertion
- Shadow corrosion
- Other anomalies
INTEGRAL EFFECTS
Maximum fuel temperature
Maximum fuel rod pressure
Maximum cladding strain (PCMI)
Maximum cladding corrosion and hydrogen content
LOCA, coolability after ballooning
LOCA, zero ductility limit
RIA, fuel failure limit
RIA, pressure shock
Power - cooling mismatch transients
EXISTING OR PLANNED HALDEN TESTS ON:
- Fuel temperature
- FGR, steady state
- FGR, transients
- Cladding creep-out
- Fuel swelling
- Cladding elongation
- Diameter PCMI
- Rod overpressure
- Power ramps
- Cladding corrosion
FUEL SURVEILLANCE PROGRAMS ON:
- Dimensional changes
- Corrosion
- Destructive PIE
- Planned LOCA tests Halden (single rod)
- Proposed bundle test in France
INTENDED ACTIVITIES AT EG.:
- ANL, USA
- JAERI, Japan ongoing
- CEA / IPSN, France
- Russia + VVER users
EXISTING OR PLANNED TESTS IN:
- CABRI-Na, France
- NSRR, Japan
PLANNED CABRI WATER LOOP TESTS
EXISTING TESTS AT HALDEN IN SPECIFIC TEST CONDITIONS AND FUEL SURVEILLANCE PROGRAMS
Existing or ongoing tests
Intended test